Table 8.46. Primary System Contaminated Equipment

Material	Quantity	Estimated Surface Radiation Level	Estimated Volume (ft^3)
Reactor coolant pump impellers	4	5 R/hr[a]	3000
Reactor coolant pump motors	4	10 mR/hr	9200
Underwater vacuum	1	300 mR/hr	30
Underwater cutting machines	2	300 mR/hr	10
Tools and grapples	-	50 mR/hr	100
Core filter support structure	8000 lb	100 mR/hr	100

[a] In seal area.

Table 8.47. Contaminated Equipment--Minimum and Maximum Waste Generation

	Minimum Generation		Maximum Generation	
Factor	Equipment	Mirror Insulation	Equipment	Mirror Insulation
Original volume[a] (ft^3)	3,450	15,000	36,000	15,000
Process used	Sectioning	Baling	None	None
Volume reduction factor[b]	2	2.5	None	None
Package type	LSA box	70-ft^3 liner[d]	LSA box[c]	LSA box
Number of packages	33	86	450	188

[a] From Table 8.43; minimum equipment consists of 3000 ft^3 plus 450 ft^3 for defueling; maximum equipment consists of 24000 ft^3 from RB plus 12000 ft^3 from defueling.

[b] Assumes 50 percent of volume can be sectioned with volume reduction factor of 2.

[c] LSA box assumed to have 80-ft^3 capacity. If 120-ft^3-capacity LSA liners are used, number of packages is reduced by 33 percent.

[d] Each liner contains 0.7 Ci; could require shielded shipment.

Table 8.48. Reactor Vessel--Irradiated Hardware

Material	Quantity	Estimated Volume (ft^3)	Estimated Surface Radiation Level (mR/hr)
Thermal insulation from reactor head	16 sections	120	300
Diaphragm over seal plate	1	8	50
Stud and nut fragments	15	24	50
Upper plenum assembly fragments	150	1200	50
Core support assembly	225	1300	50
Electric cables and coolant lines	200	300	300
Orifice rods Control rods Burnable poison rods Axial power shaping rods	177		Assumed to be removed with fuel
Control & shaping rod-drive mechanisms	69	52	50
Control & shaping rod-lead screws	69	67	50
Control & shaping rod-lead stators	69	100	300

The alternatives considered provide a basis for bounding the quantities of packaged irradiated hardware that could be generated. The bounding conditions considered were as follows:

- Maximum Packaged Waste. The items listed in Table 8.48, with radiation levels requiring shipment in a shielded overpack and no volume reduction.
- Minimum Packaged Waste. The items listed in Table 8.48 with the radiation levels shown plus volume reduction by a factor of 2 through sectioning and disassembly.

The quantities and characteristics of waste packages under these bounding conditions are shown in Table 8.49.

8.3.3 Details of Methods and Facilities

The methods used to handle and package solid materials will be selected and performed using procedures and techniques that (1) maintain site personnel exposure at "as low as reasonably achievable" (ALARA) levels, (2) minimize the volumes of packaged waste, and (3) minimize the probability of a handling accident which could breach the integrity of the disposable container in which the waste is packaged. The methods currently used to package and handle AFHB trash would be continued.

Drums containing compacted trash have been handled on 16-ft^2 pallets that hold four drums each and are moved with forklifts. Palletization of drums reduces the handling time and provides a stable means of stacking drums in an interim storage area to await shipment. The larger LSA boxes also have been handled with forklifts. Palletized drums and LSA boxes are stacked three high in an interim storage area to await transfer to a transport vehicle.

Table 8.49. Irradiated Hardware--Minimum and Maximum Waste Generation

Factor	Minimum Generation	Maximum Generation
Radiation levels less than 200 mR/hr[a]		
Original volume (ft^3)	2700	NA[b]
Process used	Dissassembly	NA
Volume reduction factor	2	NA
Package type	LSA box	NA
Number of packages	31	
Radiation levels greater than 200 mR/hr[a]	510	3300
Original volume	None	None
Process used	None	None
Volume reduction factor	None	None
Package type	70-ft^3 cask	70-ft^3 cask
Number of cask shipments[c]	15	105

[a] At surface of equipment.

[b] NA = not applicable. (For the maximum generation conditions it is assumed that all irradiated hardware would have radioactivity levels in excess of 200 mR/hr.)

[c] 70-ft^3 capacity cask and 50 percent packaging efficiency.

8.3.3.1 Trash

The practices and techniques used to collect, package, and handle trash from decontamination of the reactor building and from defueling will be essentially the same as those currently used for AFHB trash.

Personnel anticontamination clothing will be accumulated in the containment service building. Other dry waste, both compactible and noncompactible, will be accumulated at several locations in the reactor building. The accumulated material will be wrapped in polyethylene and removed from the reactor building through the equipment hatch. This bagged waste then will be transferred to the packaging area. It will be segregated, i.e., compactible versus noncompactible, and then packaged. Compactible trash will continue to be processed through the existing drum compactor to achieve a volume reduction factor of 5 and placed in 55-gallon drums. AFHB noncompactible trash has been packaged in 80-ft^3-capacity wooden LSA boxes. In the future, these 80-ft^3 wooden boxes and 120-ft^3-capacity metal LSA boxes will be used to package noncompactible trash and contaminated equipment. After packaging, materials will be moved to the low-level storage area via forklift truck to await shipment. Drums will be moved and stored on four-drum pallets.

The minimum and maximum quantities of trash that could be generated are listed in Table 8.50. As shown, incineration of combustible trash could reduce the number of drums containing trash by a factor of 10. No decision has been made on the use of incineration, and a comparison of this technique relative to compaction is summarized in Table 8.51. As illustrated in this table, both alternatives have advantages and disadvantages.

Table 8.50. Packaged Trash Summary

Factor	Compactible, Combustible Trash With Incineration	 Without Incineration	Compactible, Noncombustible Trash	Noncompactible, Noncombustible Trash
Minimum Case				
Original volume (ft^3)	168,000	168,000	56,000	82,000
Package type	Drum	Drum	Drum	LSA box
Number of packages	480	4,800	1,600	1,025[a]
Average Ci/package	2	0.2	0.2	0.1
Maximum surface radiation level/package[e]	1.6 R/hr	160 mR/hr	160 mR/hr	80 mR/hr
Maximum Case				
Original volume (ft^3)	341,000	341,000	114,000	143,000
Package type	Drum	Drum	Drum	LSA box
Number of packages	975	9,740	3,260	1,790[a]
Average Ci/package	2[b]	0.2[c]	0.2	0.1[d]
Maximum surface radiation level/package[e]	1.6 R/hr	160 mR/hr	160 mR/hr	80 mR/hr

[a] Based on 80-ft^3 capacity. If 120-ft^3-capacity LSA liners are used, number of packages would be reduced by 33 percent.

[b] From Table 8.43, trash contains 0.006 Ci/ft^3; at 35 ft^3/drum compacted, each drum contains 0.2 Ci.

[c] Incinerator ash drums contain equivalent of 350 ft^3 of trash, or 10 times the amount in drums containing compacted trash.

[d] From Table 8.43, this trash contains 0.001 Ci/ft^3; at 80 ft^3 per LSA box, each box contains about 0.1 Ci.

[e] Radiation level estimates based on 0.8 R/hr per Ci.

Table 8.51. Comparison of Incineration Versus Compaction of Combustible Trash

Factor	Minimum Trash Volumes		Maximum Trash Volumes	
	Incineration	Compaction	Incineration	Compaction
Number of drums	480	4,800	975	9,740
Number of shipments	34	40	70	81
Plant storage space (ft^2)[a]	1,440	14,400	2,900	29,000
Disposal site use (ft^3)[b]	7,200	72,000	14,600	146,000
Occupational exposure (person-rem)				
Packaging, handling, transportation	35	45	67	84
Releases to atmosphere (Ci)	1.5×10^{-4}	1×10^{-4}[c]	3×10^{-4}	2×10^{-4}[c]
Time to implement (years)	2	0	2	0
Capital cost ($ millions)	5.6	0	5.6	0
Operational cost ($ millions)	0.43-0.65	0.49-0.61	0.88-1.3	0.98-1.2

[a] Drums stacked two high; 3 ft^2/drum.

[b] 15 ft^3/drum.

[c] Based on building vent system with conservative HEPA decontamination factor of 10^3.

8.3.3.2 Contaminated Equipment

If contaminated equipment cannot be decontaminated in-place, it will be disassembled and moved with forklifts and cranes to a package-preparation area in the containment service building. Equipment that can be decontaminated using special techniques will be moved to separate decontamination stations. Equipment that must be disposed of as waste will be separated by size for compatibility with LSA boxes and radioactivity level.

The smaller equipment that will fit in LSA boxes will be packaged using "hands-on" contact methods. Some equipment may be wrapped in polyethylene prior to placement in the LSA box or while awaiting packaging in the laydown area. Individual boxes will be loaded within shielded enclosures as required to minimize exposure to site personnel. Administrative controls also will be placed on the surface dose for individual boxes.

Larger items of equipment and materials that can be readily reduced in volume will also be handled and packaged using "hands-on" methods such as use of power saws.

The volume reduction techniques described in Section 8.3.2.2, (disassembly, sectioning, and baling) will be used to the extent practicable on a case-by-case basis.

For large items like the reactor coolant pumps and motors, emphasis will be on in-place decontamination. If in-place decontamination is not successful, these units will be disassembled to the extent practicable and removed from the reactor building through the equipment hatch to an onsite decontamination area. If further decontamination is not possible, they will be packaged in special containers for offsite shipment and disposal.

Large crates would be built for equipment that could not be readily sectioned for placement in LSA boxes. After packaging, the LSA boxes would be transferred to storage to await shipment. Large crates would be shielded and left in place to await shipment.

The minimum and maximum quantities of packaged contaminated equipment that could be generated are shown in Table 8.52. In this table, the minimum and maximum unpackaged volumes given in Table 8.43 are used, and it has been assumed that 50 percent of this equipment is compatible with techniques that will reduce packaged volume by a factor of two.

Table 8.52. Packaged Contaminated Equipment Summary

Factor	Intact Equipment	Sectioned Equipment[a]	Mirror Insulation
Minimum Generation			
Original Volume (ft^3)	1,725	1,725	15,000[c]
Package type	70-ft^3 liner	70-ft^3 liner	70-ft^3 liner
Number of packages	25	13	86
Average Ci/package	0.7	1.3	0.7
Package surface radiation level	500 mR/hr[d]	1000 mR/hr[d]	500 mR/hr[d]
Maximum Generation			
Original volume (ft^3)	18,000	18,000	15,000
Package type	LSA box[b]	LSA box	LSA box
Number of packages	225	113	188
Average Ci/package	0.2	0.4	0.3
Package surface radiation level	160 mR/hr	320 mR/hr	240 mR/hr

[a] Sectioning reduces volume by factor of 2.

[b] LSA box assumed to have 80-ft^3 capacity. If 120-ft^3-capacity LSA liners are used, number of packages is reduced by 33 percent and curie content and radiation level are increased by 50 percent.

[c] Minimum generation based on baling with volume reduction factor of 2.5.

[d] Packages with these radiation levels could require shielded shipment.

8.3.3.3 Irradiated Hardware

The irradiated hardware listed in Table 8.48 could be packaged in steel containers within the reactor building and transferred through the equipment hatch. The characteristics of some of this irradiated hardware warrant special handling considerations as discussed below.

Control Rods and Control Rod Lead Screws

The exact extent of core damage is unknown, but some of these components may be fused to fuel assemblies to the extent they cannot be readily separated. Such components may have to be handled together with the damaged fuel assembly to which they are attached.

Control Rod Drive Mechanisms

Each control rod drive mechanism is about 4½ inches in diameter and 17 ft long, and the stator sections are 9½ inches in diameter and 21 inches long. These drive mechanisms are normally attached to the reactor pressure vessel head. Because of the expected high radiation dose rates from internal surfaces of these mechanisms, the whole assembly motor and pressure boundary will be cut off from each mechanism and handled underwater or dismantled and packaged for shipment if radiation levels permit.

The other items of irradiated hardware are expected to have relatively low radiation levels, below 1 R/hr. This hardware will be sectioned in the containment service building and packaged in LSA boxes or in containers compatible with shielded shipment as required.

The bounding conditions shown in Table 8.49 are the staff's estimate of the best- and worst-case requirements for packaged irradiated hardware.

8.3.4 Effluents and Releases to the Environment

The nature and impacts of releases to the environment that could occur during solid waste packaging and handling under normal conditions and under abnormal conditions or accidents are discussed below.

8.3.4.1 Normal Operations

Generally, airborne effluents arising from waste packaging operations are vented to the plant exhaust system. Under normal conditions, the operations involved in packaging and handling of contaminated equipment and irradiated hardware would not result in effluent releases to the environment. Compaction of trash could result in release of radioactive effluents to the plant exhaust system, and the releases that could arise from these operations are discussed below.

Trash Compaction

The radionuclide content of trash consists of surface contamination, and a portion of this material in the form of particulates could be released to the plant vent system during compaction. The gross activity of each drum is conservatively estimated to be about 0.2 Ci. The amounts (curies) of major radionuclides that could be released during compaction of trash for a fractional release of 10^{-4}, or 0.01 percent of drum radionuclide to the building atmosphere content, are shown in Table 8.53.

Table 8.53. Estimated Radionuclide Releases to Building Atmosphere during Trash Compaction

	Amount Released to Building Atmosphere (Ci/drum)		Total Amount Released to Building Atmosphere (Ci)[a]	
Radionuclide	AFHB Trash	Reactor Building Trash	AFHB Trash	Reactor Building Trash
Cs-137	1.6×10^{-5}	1.8×10^{-5}	5.3×10^{-2}	1.7×10^{-1}
Cs-134	2.8×10^{-6}	1.6×10^{-6}	9.2×10^{-3}	1.6×10^{-2}
Sr-90	1.2×10^{-6}	2.0×10^{-7}	3.9×10^{-3}	1.9×10^{-3}
Sr-89	4.0×10^{-7}	-	1.3×10^{-3}	-

[a] Releases before HEPA filters. The HEPA filter on the building exhaust system would also reduce these values by a factor of 10^3.

Trash Incineration

If an incinerator is used for combustible trash, the effluents released during normal operations depend on the method of combustion, the combustion chamber design, and the operational characteristics of the off-gas cleanup system. For well-designed commercial incinerators, up to 98 percent of the ash containing nonvolatile radionuclides can be retained in the combustion chamber. The off-gas cleanup system typically consists of a wet scrubber, followed by a HEPA filter, followed by a volatile radionuclide absorption system. This cleanup train typically has an off-gas decontamination factor of 10^6 to 10^7. To quantify reactor building trash incinerator particulate effluents, it is assumed by the staff that 85 percent of the nonvolatile radionuclides would be retained in the ash in the combustion chamber and that the off-gas cleanup system had a decontamination factor of 10^6. Based on these assumptions, the estimated radionuclide releases per hour of incinerator operation with a 33-ft^3-per-hour trash feed are presented in Table 8.54. The amounts shown would be released directly to the environment.

Table 8.54. Estimated Radionuclide Releases to the Environment during Trash Incineration

Radionuclide	Amount Released (Ci/hr)	Total Amount Released (Ci)[a]
Cs-137	2.3×10^{-8}	2.4×10^{-4}
Cs-134	4.2×10^{-9}	4.3×10^{-5}
Sr-90	1.8×10^{-9}	1.9×10^{-5}
Sr-89	5.9×10^{-10}	6.1×10^{-6}

[a] Based on the fractional release of a maximum 341,000 ft^3 of combustible trash from Table 8.50, using a 33-ft^3-per-hour trash feed.

8.3.4.2 Accident Conditions

The accidents postulated during packaged-waste handling and storage can be divided into two categories: (1) storage-area fires and (2) breach of container integrity as a result of the container's being punctured or dropped during waste handling operations. The accident conditions postulated and their consequences in terms of effluent releases are discussed below.

Storage-Area Fire

A worst-case accident would arise from a fire in the storage area for waste with low levels of radioactivity. The conservative conditions postulated for this accident are as follows:

- The low-level packaged waste in the storage area consists of 200 drums containing compacted trash and 40 LSA boxes of uncompacted trash.
- The fractional radionuclide release rate, in the form of respirable ash, for the combustible trash in the storage area is 0.1 volume percent.

The principal radionuclide releases arising from this postulated accident are given in Table 8.55. The gross radioactivity in compacted trash drums and LSA boxes was assumed to be 0.2 and 0.1 Ci, respectively.

Table 8.55. Estimated Radionuclide Releases to the Environment during Low-Level Waste Storage Area Fire

Radionuclide	Fraction (%) in Storage Area	Amount Released (Ci)
Cs-137	78	3×10^{-2}
Cs-134	14	6.2×10^{-3}
Sr-90	6	2.6×10^{-3}
Sr-89	2	9×10^{-4}

Packaging/Handling Accidents

The consequences of these accidents would depend on the waste type and its condition, the radioactivity and radionuclide content of the breached container, the fractional release of materials in the container, the interaction with other containers, and the area in the plant where the accident occurred.

The worst-case conditions considered for each type of packaged solid are summarized in Table 8.56.

To estimate the amounts of radionuclides released in the form of respirable particulates for each of these accidents, it was assumed that 10^{-3} of the packaged waste volume would be released for trash and mirror insulation packages. The fractional release rate assumed for immobilized incinerator ash was 10^{-5} of the package contents. The estimated releases for each of the accidents shown in Table 8.56 are presented in Table 8.57. The releases shown are based on worst-case (maximum) radionuclide content for each package type.

Table 8.56. Postulated Worst-Case Package Handling Accidents Involving Solid Waste[a]

Waste Type	Container	Accident
LSA compactible trash	Drum	Forklift penetration
Incinerator ash	Drum	Drop from crane or monorail
LSA Noncompactible trash	LSA box	Forklift penetration
Mirror insulation	70-ft^3 liner	Drop from crane or monorail

[a] All accidents assumed to occur on the loading dock, with releases directly to the environs.

Table 8.57. Estimated Accident Releases to the Environment

	Amount Released (Ci)			
Radionuclide	Compactible Trash[a]	Noncompactible Trash[b]	Mirror Insulation[c]	Incinerator Ash[d]
Cs-137	1.6×10^{-4}	7.8×10^{-5}	2.3×10^{-4}	1.6×10^{-5}
Cs-134	2.8×10^{-5}	1.4×10^{-5}	4.2×10^{-5}	2.8×10^{-6}
Sr-90	1.2×10^{-5}	$6. \times 10^{-6}$	1.8×10^{-5}	1.2×10^{-6}
Sr-89	4×10^{-6}	2×10^{-6}	6×10^{-6}	4×10^{-7}

[a] 0.2 Ci/drum.

[b] 0.1 Ci/LSA box.

[c] 0.3 Ci/70-ft^3 liner.

[d] 2 Ci/drum.

8.3.5 Environmental Impacts

8.3.5.1 Occupational Doses

Occupational doses for waste handling and packaging, through placement of packaged waste in onsite storage facilities to await shipment, are summarized in this section. Based on information in Appendix N, doses were projected for best- and worst-case conditions based on minimum and maximum waste generation volumes. These exposure estimates for AFHB, reactor building, and defueling waste are presented in Tables 8.58, 8.59, and 8.60, respectively, and are summarized in Table 8.61.

AFHB Wastes

AFHB waste (see Table 8.58) would be packaged and handled by two-person crews, and four crews would be used over the 15-month period required to perform this activity. The average exposure to each crew member over a 15-month period is estimated to be 2.8 rem, or an average of 0.6 rem per quarter. If the work were conducted over an 18-month period, the average dose would be 0.5 rem per crew member per quarter.

The expected number of additional cancer mortalities in the work force of eight persons receiving this cumulative dose of radiation would be 0.003. This means that the added probability that the average individual worker would die of cancer would be 1 in 3000. The expected number of additional genetic effects in the offspring of the work force receiving to this cumulative dose of radiation would be 0.005.

Reactor Building Wastes

Reactor building wastes (see Table 8.59) would be packaged and handled by two-member crews, and four crews would be used over the period required to complete this activity. Under minimum waste generation conditions, this activity could be performed over an 18-month period, and crew members would receive an average radiation dose of about 1.6 rem, or about 0.3 rem per quarter. Under maximum waste generation conditions, this activity could be performed over a 30-month period, and each crew member would receive a total of about 5 rem, or 0.5 rem per quarter.

For the cumulative occupational dose range of 12 to 43 person-rem, the expected number of additional cancer mortalities in the work force of eight people ranges between 0.002 and 0.006. This means that the added probability that the average individual worker would die of cancer ranges from 1 in 5000 to 1 in 1500. The expected number of additional genetic effects in the offspring of the work force exposed to this cumulative dose of radiation ranges between 0.003 and 0.01.

Defueling Wastes

Defueling wastes (Table 8.60) would be packaged and handled by two-member crews, and three crews would be used over an 18- to 36-month period. Under minimum waste generation conditions, the activity could be completed in 18 months, and each crew member would receive a total dose of about 1.4 rem, or about 0.24 rem per quarter. Under maximum waste generation conditions, the activity could be completed in 30 months, and each crew member would receive a total of about 4.7 rem, or about 0.47 rem per quarter.

Based on a maximum cumulative dose of 30 person-rem for handling and packaging of solid wastes, the expected number of additional cancer mortalities in the work force exposed to this maximum cumulative dose of radiation would be 0.004. The added probability that the average individual worker would die of cancer would be 1 in 1500. The expected number of additional genetic effects in the offspring of the work force exposed to this maximum cumulative dose of radiation would be 0.008.

Table 8.58. Estimated Occupational Radiation Doses from Handling and Packaging of AFHB Solid Wastes

Waste Form	Package Type	Unit Dose (person-mrem/ package)	Cumulative Occupational Dose (person-rem)
Trash--without incineration[a]			
Compactible	Drum	3.5	11.5
Noncompactible	LSA box	21	13
Total			25[b]
Trash--with incineration[a]			
Combustible	Drum	17	4.2
Compactible	Drum	3.5	2.8
Noncompactible	LSA box	21	13
Total			20

[a] Alternatives being considered.

[b] Rounded to two significant figures.

Table 8.59. Estimated Occupational Radiation Doses from Handling and Packaging of Reactor Building Solid Wastes

Waste Form	Package Type	Unit Dose (person-mrem/ package)	Cumulative Occupational Dose (person-rem)	
			Best Case	Worst Case
Trash--without incineration[a]				
Compactible	Drum	3.5	6	20
Noncompactible	LSA box	21	5.2	14
Subtotal			11[b]	34
Trash--with incineration[a]				
Combustible	Drum	17	2.2	7.5
Compactible	Drum	3.5	1.5	5
Noncompactible	LSA box	21	5.2	14
Subtotal			8.9	27[b]
Contaminated equipment	LSA box	21	0.8	4.7
Mirror insulation				
Low activity	LSA box	21	-	3.9
High-specific activity[c]	Liners	22	2.2	-
Totals[b]			12-14[d]	36-43[d]

[a] Alternatives being considered.

[b] Rounded to two significant figures.

[c] Same original volume reduced by compaction, which increases specific activity.

[d] Range reflects effect of incineration.

Table 8.60. Estimated Occupational Radiation Doses from Handling and Packaging of Solid Wastes from Defueling and Primary System Decontamination

Waste Form	Package Type	Unit Dose (person-mrem/package)	Cumulative Occupational Dose (person-rem)	
			Best Case	Worst Case
Trash--without incineration[a]				
Compactible	Drum	3.5	5	14
Noncompactible	LSA box	21	3.5	11
Subtotal			8.5	25
Trash--with incineration[a]				
Combustible	Drum	17	1.8	6.1
Compactible	Drum	3.5	1.2	3.5
Noncompactible	LSA box	21	3.5	11
Subtotal			6.5	20
Contaminated equipment	LSA box	21	0.1	3.1
Irradiated hardware				
Low activity	LSA box	21	0.6	-
High-specific activity	Liners	22	0.3	2.3
Totals[b]			7.5-9.5[c]	26-30[c]

[a] Alternatives being considered.

[b] Rounded to two significant figures.

[c] Range reflects effect of incineration.

Table 8.61. Summary of Estimated Occupational Radiation Doses from Handling and Packaging of Solid Wastes[a]

Waste Form	Cumulative Occupational Dose (person-rem)	
	Best Case	Worst Case
Trash--without incineration	45.0	84.0
Trash--with incineration	35.0	67.0
Contaminated equipment	1.0	9.0
Mirror insulation	2.2	3.9
Irradiated hardware	1.0	2.3
Total	39-49[b]	82-99[b]

[a] Combines estimates from Tables 8.59, 8.60, and 8.61.

[b] Range in totals reflects impact of trash incineration, rounded to two significant figures.

8.3.5.2 Offsite Doses

The dose estimates presented here are for normal releases from trash compaction and incineration. The source terms are presented in Section 8.3.4.1, Tables 8.53 and 8.54. The calculational models used to make these estimates and the interpretation of their results are described in Appendix W. The significance of these doses and their human health and environmental consequences are discussed in Section 10.3. The dose estimates to the maximum exposed individual for trash compaction of AFHB wastes are listed in Table 8.62, and the 50-mile population dose was estimated to be 1×10^{-3} person-rem. The dose estimates to the maximum exposed individual for trash compaction of reactor building wastes are listed in Table 8.63, and the 50-mile population dose was estimated to be 2×10^{-3} person-rem. The dose estimates to the maximum exposed individual for trash incineration are listed in Table 8.64, and the 50-mile population dose was estimated to be 6×10^{-3} person-rem.

8.3.5.3 Postulated Accident Effects

The types of accidents for which dose estimates are made here are the following: (1) fire in low-level storage area; (2) breach of a package containing compactable trash; (3) breach of a package containing noncompactible trash; (4) breach of a package containing mirror insulation; and (5) breach of a package containing incinerator ash. The accidents are described in Section 8.3.4.2 and their source terms are listed in Tables 8.55 and 8.57. The calculational models used to make these estimates and the interpretation of their results are described in Appendix W. The significance of these doses is discussed in Section 10.4. The dose estimates for the maximum exposed individual are in Table 8.65 for the low-level trash fire, Table 8.66 for the breach of a package containing compactable trash, Table 8.67 for the breach of a package containing noncompactable trash, Table 8.68 for the breach of a package containing mirror insulation trash, and Table 8.69 for the breach of a package containing incinerator ash.

8.3.5.4 Psychological-Socioeconomic Effects

Decontamination of the AFHB and reactor building, primary system processing, reactor defueling, and primary system decontamination involve activities that generate solid waste. Such waste can be classified as either solid materials (trash, contaminated equipment, irradiated hardware, and damaged fuel assemblies) or process solids (exchange resins, accident sludge, and evaporator bottoms).

The staff concludes that although airborne releases will pose a negligible health threat to individuals living in the vicinity of TMI, the packaging and handling of solid waste could arouse additional distress for some members of the public because of uncertainties surrounding the ultimate disposal of the waste. For local people, this concern is focused on the possibility of using the station site as a long-term storage facility and is a factor exacerbating existing uncertainty and anxiety.

Although they would have negligible offsite health consequences, the accidents depicted in the scenarios considered by the staff would aggravate existing public uncertainty, especially in local communities. The level and duration of psychological distress would be associated with the character of the initiating threat, the level of controversy, and/or the type and length of media coverage.

8.3.6 Economic Costs

Solid waste materials consist of trash, contaminated equipment, and irradiated hardware. The costs associated with managing these waste forms consist of those for treatment, conditioning, and packaging and handling. Not all waste forms require consideration of all three areas of cost. As indicated previously, some waste materials can be packaged directly without treatment or conditioning and therefore can be processed with less expense. In other cases, both treatment and conditioning are required and higher costs result. Using this basis, the staff has developed bounding costs for each of the three waste categories as shown in Table 8.70. Details supporting these cost estimates are presented in Appendix K.

Table 8.62. Dose Estimates for the Maximum Exposed Individual for Trash Compaction of AFHB Wastes

Location	Pathway	Dose (mrem)[a]		
		Total-Body	Bone	Liver
Nearest garden[b]	Inhalation	1.8×10^{-6}	2.4×10^{-5}	2.8×10^{-6}
	Ground Shine	6.2×10^{-6}	6.2×10^{-6}	6.2×10^{-6}
	Vegetable Use	1.4×10^{-4}	6.0×10^{-4}	1.4×10^{-4}
	Total	1.5×10^{-4}	6.3×10^{-4}	1.5×10^{-4}
Nearest milk goat	Inhalation	2.6×10^{-6}	9.3×10^{-6}	1.9×10^{-6}
	Ground Shine	6.0×10^{-6}	6.0×10^{-6}	6.0×10^{-6}
	Goat Milk Use	1.0×10^{-4}	7.3×10^{-4}	8.2×10^{-4}
	Total	1.1×10^{-4}	7.5×10^{-4}	8.3×10^{-4}
Nearest cow and garden	Inhalation	2.0×10^{-6}	2.6×10^{-5}	3.1×10^{-6}
	Ground Shine	9.2×10^{-6}	9.2×10^{-6}	9.2×10^{-6}
	Vegetable Use	2.1×10^{-4}	8.9×10^{-4}	2.0×10^{-4}
	Cow Milk Use	3.4×10^{-5}	1.9×10^{-4}	1.6×10^{-4}
	Total	2.6×10^{-4}	1.1×10^{-3}	3.7×10^{-4}

[a] Doses were calculated for total-body, GI-tract, bone, liver, kidney, thyroid, lung, and skin. The maximum three-organ doses are listed in this table. Doses were calcuated for four age groups: adults, teenagers, children, and infants. The highest dose estimates for each age group are listed. The dose estimates for the nearest garden and for the nearest cow and garden locations are for children. The dose estimates for the nearest goat location are for adults for total-body and for infants for bone and liver.

[b] The basis for selecting the special locations is described in Appendix W. The actual locations are: nearest garden = 1.05 miles east-northeast, nearest milk goat = 1.02 miles north, and nearest cow and garden = 1.05 miles east.

Table 8.63. Dose Estimates for the Maximum Exposed Individual for Trash Compaction of Reactor Building Wastes

Location	Pathway	Dose (mrem)[a]		
		Total-Body	Bone	Liver
Nearest garden[b]	Inhalation	5.1×10^{-6}	1.9×10^{-5}	8.4×10^{-6}
	Ground Shine	1.7×10^{-5}	1.7×10^{-5}	1.7×10^{-5}
	Vegetable Use	1.3×10^{-4}	6.4×10^{-4}	4.1×10^{-4}
	Total	1.5×10^{-4}	6.8×10^{-4}	4.4×10^{-4}
Nearest milk goat	Inhalation	4.6×10^{-6}	8.6×10^{-6}	5.6×10^{-6}
	Ground Shine	1.7×10^{-5}	1.7×10^{-5}	1.7×10^{-5}
	Goat Milk Use	2.8×10^{-4}	2.0×10^{-3}	2.4×10^{-3}
	Total	3.0×10^{-4}	2.0×10^{-3}	2.4×10^{-3}
Nearest cow and garden	Inhalation	5.7×10^{-6}	2.1×10^{-5}	9.2×10^{-6}
	Ground Shine	2.5×10^{-5}	2.5×10^{-5}	2.5×10^{-5}
	Vegetable Use	1.9×10^{-4}	9.4×10^{-4}	6.1×10^{-4}
	Cow Milk Use	1.1×10^{-4}	4.8×10^{-4}	4.7×10^{-4}
	Total	3.3×10^{-4}	1.5×10^{-3}	1.1×10^{-3}

[a] Doses were calculated for total-body, GI-tract, bone, liver, kidney, thyroid, lung, and skin. The maximum three-organ doses are listed in this table. Doses were calcuated for four age groups: adults, teenagers, children, and infants. The highest dose estimates for each age group are listed. The dose estimates for the nearest garden and for the nearest cow and garden locations are for children. The dose estimates for the nearest milk goat location are for adults for total-body and for infants for bone and liver.

[b] The basis for selecting the special locations is described in Appendix W. The actual locations are: nearest garden = 1.05 miles east-northeast, nearest milk goat = 1.02 miles north, and nearest cow and garden = 1.05 miles east.

Table 8.64. Dose Estimates for the Maximum Exposed Individual for Releases Made due to Trash Incineration

Location	Pathway	Dose (mrem)[a]		
		Total-Body	Bone	Liver
Nearest garden[b]	Inhalation	8.7×10^{-6}	1.2×10^{-4}	1.3×10^{-5}
	Ground Shine	2.9×10^{-5}	2.9×10^{-5}	2.9×10^{-5}
	Vegetable Use	6.8×10^{-4}	2.9×10^{-3}	6.3×10^{-4}
	Total	7.2×10^{-4}	3.0×10^{-3}	6.7×10^{-4}
Nearest milk goat	Inhalation	1.2×10^{-5}	4.5×10^{-5}	8.6×10^{-6}
	Ground Shine	2.7×10^{-5}	2.7×10^{-5}	2.7×10^{-5}
	Goat Milk Use	4.8×10^{-4}	3.4×10^{-3}	3.7×10^{-3}
	Total	5.2×10^{-4}	3.5×10^{-3}	3.7×10^{-3}
Nearest cow and garden	Inhalation	9.6×10^{-6}	1.3×10^{-4}	1.4×10^{-5}
	Ground Shine	4.2×10^{-5}	4.2×10^{-5}	4.2×10^{-5}
	Vegetable Use	9.9×10^{-4}	4.3×10^{-3}	9.3×10^{-4}
	Cow Milk Use	1.6×10^{-4}	8.7×10^{-4}	7.2×10^{-4}
	Total	1.2×10^{-3}	5.3×10^{-3}	1.7×10^{-3}

[a] Doses were calculated for total-body, GI-tract, bone, liver, kidney, thyroid, lung, and skin. The maximum three-organ doses are listed in this table. Doses were calcuated for four age groups: adults, teenagers, children, and infants. The highest dose estimates for each age group are listed. The dose estimates for the nearest garden and for the nearest cow and garden locations are for children. The dose estimates for the nearest milk goat location are for adults for total-body and for infants for bone and liver.

[b] The basis for selecting the special locations is described in Appendix W. The actual locations are: nearest garden = 1.05 miles east-northeast, nearest milk goat = 1.02 miles north, and nearest cow and garden = 1.05 miles east.

Table 8.65. Dose Estimates to the Maximum Exposed Individual Caused by a Low-Level Storage Area Fire

Location	Pathway	Dose (mrem)[a]		
		Total-Body	Bone	Liver
Nearest garden[b]	Inhalation	2.0×10^{-2}	2.7×10^{-1}	2.9×10^{-2}
	Ground Shine	2.2×10^{-1}	2.2×10^{-1}	2.2×10^{-1}
	Vegetable Use	5.4	24	4.7
	Total	5.6	24	4.9
Nearest milk goat	Inhalation	3.1×10^{-2}	1.2×10^{-1}	2.1×10^{-2}
	Ground Shine	2.2×10^{-1}	2.2×10^{-1}	2.2×10^{-1}
	Goat Milk Use	3.8	27	30
	Total	4.1	27	30
Nearest cow and garden	Inhalation	1.4×10^{-2}	1.9×10^{-1}	2.0×10^{-2}
	Ground Shine	2.2×10^{-1}	2.2×10^{-1}	2.2×10^{-1}
	Vegetable Use	5.4	24	4.7
	Cow Milk Use	8.2×10^{-1}	4.4	3.6
	Total	6.5	29	8.5

[a] Doses were calculated for total-body, GI-tract, bone, liver, kidney, thyroid, lung, and skin. The maximum three-organ doses are listed in this table. Doses were calcuated for four age groups: adults, teenagers, children, and infants. The highest dose estimates for each age group are listed. The dose estimates for the nearest garden and for the nearest cow and garden locations are for children. The dose estimates for the nearest milk goat location are for adults for total-body and for infants for bone and liver.

[b] The basis for selecting the special locations is described in Appendix W. The actual locations are: nearest garden = 1.05 miles east-northeast, nearest milk goat = 1.02 miles north, and nearest cow and garden = 1.05 miles east.

Table 8.66. Dose Estimates to the Maximum Exposed Individual Caused by Breaching a Package Containing Compactible Trash

Location	Pathway	Dose (mrem)[a]		
		Total-Body	Bone	Liver
Nearest garden[b]	Inhalation	9.6×10^{-5}	1.3×10^{-3}	1.5×10^{-4}
	Ground Shine	1.1×10^{-3}	1.1×10^{-3}	1.1×10^{-3}
	Vegetable Use	2.5×10^{-2}	1.1×10^{-1}	2.5×10^{-2}
	Total	2.6×10^{-2}	1.1×10^{-1}	2.6×10^{-2}
Nearest milk goat	Inhalation	1.5×10^{-4}	5.4×10^{-4}	1.1×10^{-4}
	Ground Shine	1.1×10^{-3}	1.1×10^{-3}	1.1×10^{-3}
	Goat Milk Use	1.9×10^{-2}	1.3×10^{-1}	1.5×10^{-1}
	Total	2.0×10^{-2}	1.3×10^{-1}	1.5×10^{-1}
Nearest cow and garden	Inhalation	6.6×10^{-5}	8.7×10^{-4}	1.0×10^{-4}
	Ground Shine	1.1×10^{-3}	1.1×10^{-3}	1.1×10^{-3}
	Vegetable Use	2.5×10^{-2}	1.1×10^{-1}	2.5×10^{-2}
	Cow Milk Use	4.1×10^{-3}	2.2×10^{-2}	1.9×10^{-2}
	Total	3.0×10^{-2}	1.3×10^{-1}	4.5×10^{-2}

[a] Doses were calculated for total-body, GI-tract, bone, liver, kidney, thyroid, lung, and skin. The maximum three-organ doses are listed in this table. Doses were calcuated for four age groups: adults, teenagers, children, and infants. The highest dose estimates for each age group are listed. The dose estimates for the nearest garden and for the nearest cow and garden locations are for children. The dose estimates for the nearest milk goat location are for adults for total-body and for infants for bone and liver.

[b] The basis for selecting the special locations is described in Appendix W. The actual locations are: nearest garden = 1.05 miles east-northeast, nearest milk goat = 1.02 miles north, and nearest cow and garden = 1.05 miles east.

Table 8.67. Dose Estimates to the Maximum Exposed Individual Caused by Breaching a Package Containing Noncompactible Trash

Location	Pathway	Dose (mrem)[a]		
		Total-Body	Bone	Liver
Nearest garden[b]	Inhalation	4.8×10^{-5}	6.3×10^{-4}	7.2×10^{-5}
	Ground Shine	5.4×10^{-4}	5.4×10^{-4}	5.4×10^{-4}
	Vegetable Use	1.3×10^{-2}	5.5×10^{-2}	1.3×10^{-2}
	Total	1.4×10^{-2}	5.5×10^{-2}	1.3×10^{-2}
Nearest milk goat	Inhalation	7.4×10^{-5}	2.7×10^{-4}	5.3×10^{-5}
	Ground Shine	5.4×10^{-4}	5.4×10^{-4}	5.4×10^{-4}
	Goat Milk Use	9.5×10^{-3}	6.6×10^{-2}	7.4×10^{-2}
	Total	1.0×10^{-2}	6.7×10^{-2}	7.5×10^{-2}
Nearest cow and garden	Inhalation	3.3×10^{-5}	4.4×10^{-4}	5.0×10^{-5}
	Ground Shine	5.4×10^{-4}	5.4×10^{-4}	5.4×10^{-4}
	Vegetable Use	1.3×10^{-2}	5.4×10^{-2}	1.2×10^{-2}
	Cow Milk Use	2.1×10^{-3}	1.1×10^{-2}	9.3×10^{-3}
	Total	1.6×10^{-2}	6.6×10^{-2}	2.2×10^{-2}

[a] Doses were calculated for total-body, GI-tract, bone, liver, kidney, thyroid, lung, and skin. The maximum three-organ doses are listed in this table. Doses were calcuated for four age groups: adults, teenagers, children, and infants. The highest dose estimates for each age group are listed. The dose estimates for the nearest garden and for the nearest cow and garden locations are for children. The dose estimates for the nearest milk goat location are for adults for total-body and for infants for bone and liver.

[b] The basis for selecting the special locations is described in Appendix W. The actual locations are: nearest garden = 1.05 miles east-northeast, nearest milk goat = 1.02 miles north, and nearest cow and garden = 1.05 miles east.

Table 8.68. Dose Estimates to the Maximum Exposed Individual Caused by Breaching a Package Containing Trash Mirror Insulation

Location	Pathway	Dose (mrem)[a]		
		Total-Body	Bone	Liver
Nearest garden[b]	Inhalation	1.4×10^{-4}	1.9×10^{-3}	2.1×10^{-4}
	Ground Shine	1.6×10^{-3}	1.6×10^{-3}	1.6×10^{-3}
	Vegetable Use	3.8×10^{-2}	1.6×10^{-1}	3.6×10^{-2}
	Total	4.0×10^{-2}	1.6×10^{-1}	3.8×10^{-2}
Nearest milk goat	Inhalation	2.2×10^{-4}	8.1×10^{-4}	1.6×10^{-4}
	Ground Shine	1.6×10^{-3}	1.6×10^{-3}	1.6×10^{-3}
	Goat Milk Use	2.8×10^{-2}	1.9×10^{-1}	2.2×10^{-1}
	Total	3.0×10^{-2}	1.9×10^{-1}	2.2×10^{-1}
Nearest cow and garden	Inhalation	9.8×10^{-5}	1.3×10^{-3}	1.5×10^{-4}
	Ground Shine	1.6×10^{-3}	1.6×10^{-3}	1.6×10^{-3}
	Vegetable Use	3.8×10^{-2}	1.6×10^{-1}	3.6×10^{-2}
	Cow Milk Use	6.0×10^{-3}	3.3×10^{-2}	2.7×10^{-2}
	Total	4.6×10^{-2}	2.0×10^{-1}	6.5×10^{-2}

[a] Doses were calculated for total-body, GI-tract, bone, liver, kidney, thyroid, lung, and skin. The maximum three-organ doses are listed in this table. Doses were calcuated for four age groups: adults, teenagers, children, and infants. The highest dose estimates for each age group are listed. The dose estimates for the nearest garden and for the nearest cow and garden locations are for children. The dose estimates for the nearest milk goat location are for adults for total-body and for infants for bone and liver.

[b] The basis for selecting the special locations is described in Appendix W. The actual locations are: nearest garden = 1.05 miles east-northeast, nearest milk goat = 1.02 miles north, and nearest cow and garden = 1.05 miles east.

Table 8.69. Dose Estimates for the Maximum Exposed Individual Caused by Breaching a Package Containing Incinerator Ash

Location	Pathway	Dose (mrem)[a]		
		Total-Body	Bone	Liver
Nearest garden[b]	Inhalation	9.6×10^{-6}	1.3×10^{-4}	1.5×10^{-5}
	Ground Shine	1.1×10^{-4}	1.1×10^{-4}	1.1×10^{-4}
	Vegetable Use	2.5×10^{-3}	1.1×10^{-2}	2.5×10^{-3}
	Total	2.6×10^{-3}	1.1×10^{-2}	2.6×10^{-3}
Nearest milk goat	Inhalation	1.5×10^{-5}	5.4×10^{-5}	1.1×10^{-5}
	Ground Shine	1.1×10^{-4}	1.1×10^{-4}	1.1×10^{-4}
	Goat Milk Use	1.9×10^{-3}	1.3×10^{-2}	1.5×10^{-2}
	Total	2.0×10^{-3}	1.3×10^{-2}	1.5×10^{-2}
Nearest cow and garden	Inhalation	6.6×10^{-6}	8.7×10^{-5}	1.0×10^{-5}
	Ground Shine	1.1×10^{-4}	1.1×10^{-4}	1.1×10^{-4}
	Vegetable Use	2.5×10^{-3}	1.1×10^{-2}	2.5×10^{-3}
	Cow Milk Use	4.1×10^{-4}	2.2×10^{-3}	1.9×10^{-3}
	Total	3.0×10^{-3}	1.3×10^{-2}	4.5×10^{-3}

[a] Doses were calculated for total-body, GI-tract, bone, liver, kidney, thyroid, lung, and skin. The maximum three-organ doses are listed in this table. Doses were calcuated for four age groups: adults, teenagers, children, and infants. The highest dose estimates for each age group are listed. The dose estimates for the nearest garden and for the nearest cow and garden locations are for children. The dose estimates for the nearest milk goat location are for adults for total-body and for infants for bone and liver.

[b] The basis for selecting the special locations is described in Appendix W. The actual locations are: nearest garden = 1.05 miles east-northeast, nearest milk goat = 1.02 miles north, and nearest cow and garden = 1.05 miles east.

Table 8.70. Cost Estimates for Solid Materials Processing (thousands of dollars)

Waste Type	Best Case	Worst Case
Trash	650[a]	6900[b]
Contaminated equipment	50	150
Irradiated hardware	20	80
Total	720	7130

[a] Includes major capital cost for compactor of $180,000.

[b] Includes major capital cost for compaction and incineration facilities of $5,780,000.

8.4 FUEL ASSEMBLIES AND CORE DEBRIS

As the intact and damaged fuel assemblies and core debris are removed from the reactor vessel (see Sec. 6.4 for description of this process), they will be placed in transfer containers and moved from the reactor building to the AFHB for temporary storage.

The characteristics of the removed fuel and debris, the alternatives considered for their packaging and handling, and the environmental impacts of these operations are considered in this section.

8.4.1 Status and Specific Considerations

8.4.1.1 Efforts and Practices to Date

As was previously discussed, reactor defueling, and thus fuel canning (placement in transfer containers) and transfer, is not expected to begin until 1983.

The primary effort to date has consisted of scoping studies assessing the potential status of the fuel and available options for packaging and handling the fuel after removal from the reactor. The Allied-General Nuclear Services study of September 1980 is the basis for a number of the options discussed herein and should be referred to for specific details.[8]

8.4.1.2 Projected Requirements

The TMI-2 core, as loaded, contained 177 fuel assemblies. Each assembly was about 8.5 inches square and 170 inches long. It is uncertain what the condition of the fuel will be when removed from the reactor vessel; the staff assumes that the fuel will be in any one of three configurations as follows:

- Intact--intact, but in some cases weakened, and probably bowed in the upper regions of the assemblies.
- Fused Sections--portions of adjacent fuel assemblies fused to each other such that they will have to be physically separated prior to placement in transfer containers (cans).
- Core Debris--consists of two types: relatively large pieces that can be mechanically handled, and smaller pieces that will have to be vacuumed and filtered prior to canning.

It is necessary that alternative plans be made to package and handle the range of above described conditions. In each case, it is likely that the fuel will require canning to provide containment and structural integrity and to prevent spread of contamination during the steps involved in onsite storage and handling.

The potential packaging and handling problems resulting from the physical shape of the fuel may be offset to some extent, by the low burnup of the fuel and the aging of the fuel that will have occurred prior to removal from the reactor. This fuel will have low thermal decay heat levels.

Based on the range of fuel conditions anticipated, the best-case condition is estimated to consist of 50 units (cans) of damaged fuel assemblies (both intact but weakened and separated sections), and six cans of debris, with the remainder of the fuel assemblies undamaged and capable of being handled as ordinary spent fuel. The worst case is estimated to be 177 cans of damaged fuel assemblies plus six additional cans of debris. It is likely that all fuel assemblies will have to be conservatively handled, even if they appear to be undamaged, and placed in a transfer container upon defueling.

8.4.2 Alternative Methods Considered

The alternatives considered for packaging and handling (transfer to storage) of the fuel, based on its condition, are described below.

8.4.2.1 Intact Fuel Assemblies

Fuel elements that are structurally sound, can bear their own weight, are not distorted, and retain the integrity of their cladding can be handled as normal spent fuel and transferred directly to pool storage without prior canning.

Intact elements that are structurally weakened, and possibly perforated in the upper region of the assembly, will require canning immediately upon defueling. The can would provide both vertical and lateral support of the fuel assembly as it is being moved to the AFHB and prevent spread of radioactive contamination.

Because it may not be possible to determine structural soundness of the assembly upon removal from the reactor vessel, it may be necessary to can all units.

There are two alternative routes for transfer of the fuel from the reactor building to the AFHB. The more direct and simpler route is movement directly between the buildings using the fuel transfer carriage/upender and the fuel transfer tube with the AFHB storage pool as the end-point. Because of the dimensions of the tube, transfer by this alternative would limit the size of the package to 11.48 square inches cross section (or 15.25 inches in diameter for cylindrical packages by 15.1 ft long).[1]

The second transfer route involves moving the packages through the containment hatch, external to the reactor, and back into the spent fuel pool for interim storage. This routing involves designing specialized transfer packages and equipment and entails greater operational complexity. It therefore would be appropriate only if canned fuel assemblies (or sections) were of larger dimensions than could traverse the fuel transfer tube.

Interim wet storage in the spent fuel storage pool pending packaging and offsite disposal is anticipated. Dry vault or caisson storage in a new onsite facility is another alternative for interim storage. These options are described in more detail in Section 9.

8.4.2.2 Fused Sections

Where the fuel assemblies (or sections) are fused together in a configuration that does not fit into the can used for intact assemblies, the following alternatives are available upon removal of the fuel from the reactor:

- Use of a larger transporter can or "bucket" for the fused sections and route the can through the containment hatch (see Sec. 8.4.2.1).
- Mechanically separate the larger pieces into sections small enough to be loaded into the fuel cans that can be moved via the upender through the fuel transfer tube.

Interim storage alternatives are the same as for intact assemblies.

8.4.2.3 Core Debris

The handling and packaging of the core debris potentially will require the greatest use of auxiliary equipment and the maximum number of operational steps. The remote removal of fuel debris from the core and loading into a transporter can is expected to be done in two steps as follows:[1]

- Mechanical tongs will be used to remove the larger pieces and place them in a container acting as a receiving "bucket".
- A vacuuming system will be used to remove the small pieces and fuel fines, filter, and segregate the pieces by size. These pieces will then be canned in the same manner as the larger ones.

8.4.3 Details of Methods and Facilities

The procedures and equipment used to handle and package the fuel will be intended to ensure that (1) site personnel exposure is maintained at ALARA levels and (2) the probability of a handling accident is minimized.

The following design and operational considerations will govern the implementation of any packaging and handling alternatives:

- Ideally, a single can (transfer container) design will be developed for all projected uses. This can would be sized to handle a range of fuel component sizes and also fit in the interim storage racks and potential shipping casks. The can will be designed to prevent the occurrence of criticality.
- The fuel can will be designed to be compatible with the storage pool water, if this interim storage alternative is selected. Containment to prevent dispersion of radioactivity will be ensured.
- Special procedures will be developed for fuel handling and packaging because of the uniqueness of the TMI-2 situation. Paramount will be the need to ensure that contamination does not spread during handling and canning operations. Regulatory review and approval of these procedures will be obtained.

8.4.3.1 Intact Fuel Assemblies

The likely approach for packaging and handling the intact fuel assemblies would be to place the assemblies directly into a fuel can upon removal from the reactor vessel, route the canned fuel through the fuel transfer tube, and provide interim storage in the spent fuel pool.

Following interim storage in racks in the spent fuel pool, the fuel cans will be loaded into a LWR spent fuel shipping cask for offsite transport for storage and/or reprocessing and, ultimately, disposal. These options are discussed in Section 9. A separate shipping container may be required as an additional containment barrier within the cask for offsite shipment.

8.4.3.2 Fused Sections

In the case of the fused sections, both (1) the use of larger fuel cans that would be routed outside of the reactor building and (2) the alternative of mechanically separating the fused sections to permit use of the smaller fuel cans that can be routed through the fuel transfer tube are feasible alternatives that must be considered until design and operational studies are completed.

After removal to the interim storage area, the handling procedures for cans containing fused sections will be the same as for the intact assemblies.

8.4.3.3 Core Debris

Remote handling of the core debris, either by use of mechanical tongs for larger pieces or a "vacuum" system for the small pieces, is the likely approach. After the debris are loaded into the fuel can, the can will be routed through the fuel transfer tube for interim storage in the AFHB.

8.4.4 Effluents and Releases to the Environment

The nature and impacts of the releases to the environment that could occur during the packaging and handling operations have been covered in the discussion under core defueling (Sec. 6.4).

8.4.5 Environmental Impacts

Occupational doses to workers involved in the packaging and handling operations and offsite doses to the population are treated as components of the core defueling impacts (Sec. 6.4.5).

8.4.6 Economic Costs

Capital and operating costs for the packaging and handling operations are included as part of the core defueling costs (Sec. 6.4.6).

Reference--Section 8

1. Amendment No. 10 to License No. DPR-73, U.S. Nuclear Regulatory Commission, March 12, 1980.
2. Letter from J.F. Ahearn, U.S. Nuclear Regulatory Commission to H. Dieckamp, General Public Utilities Corporation, January 12, 1981.
3. "Data Handout - TMI-2" as available to the TMI Working Group meeting held at TMI, Middletown, PA, September 23, 1980.
4. Memorandum from T.L. Gilbert to W.K. Lehto, Argonne National Laboratory, Subject: AFHB Sludge Volumes and Activities, December 9, 1980.
5. Letter from W.J. Dircks, U.S. Nuclear Regulatory Commission to C.W. Bateman, U.S. Dept. of Energy, January 7, 1981.
6. "Future EPICOR II Operation," Letter from G.K. Hovey of Metropolitan Edison Company - TMI-2 to TMI Program Office, January 13, 1981.
7. "EPICOR II Resin Solidification Procurement Specification," Letter (TLL 545) from G.K. Hovey of Metropolitan Edison Company - TMI-2 to TMI Program Office, November 17, 1980.
8. "Scoping Studies of the Alternative Options for Defueling, Packaging, Shipping, and Disposing of the TMI-2 Spent Fuel Core," Allied-General Nuclear Services, AGNS-35900-1.5-79, September 1980.

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4. Memorandum from T.L. Gilbert to W.K. Lehto, Argonne National Laboratory, Subject: AFHB Sludge Volumes and Activities, December 9, 1980.
5. Letter from W.J. Dircks, U.S. Nuclear Regulatory Commission to C.W. Bateman, U.S. Dept. of Energy, January 7, 1981.
6. "Future EPICOR II Operation," Letter from G.K. Hovey of Metropolitan Edison Company - TMI-2 to TMI Program Office, January 13, 1981.
7. "EPICOR II Resin Solidification Procurement Specification," Letter (TLL 545) from G.K. Hovey of Metropolitan Edison Company - TMI-2 to TMI Program Office, November 17, 1980.
8. "Scoping Studies of the Alternative Options for Defueling, Packaging, Shipping, and Disposing of the TMI-2 Spent Fuel Core," Allied-General Nuclear Services, AGNS-35900-1.5-79, September 1980.

9. STORAGE, TRANSPORTATION, AND DISPOSAL OF FUEL AND SOLID WASTE

The waste management activities of onsite storage of the packages of TMI-2 waste and spent fuel and the alternative offsite transportation of the waste and fuel to storage or disposal facilities are discussed in this section. The purposes of this section are to:

- Describe the waste management activities conducted to date at the TMI site subsequent to the accident.
- Provide estimates of the quantities of each type of waste package and numbers of waste shipments (shielded and unshielded).
- Discuss the regulatory and technical constraints on the waste storage, transportation, and disposal operations.
- Define the range of alternative approaches for the waste storage, transportation, and disposal options; identify the viable alternatives; and provide the details of the selected alternatives.
- Describe the effluents and releases to the environment from storage, transportation, and disposal activities.
- Determine the environmental impacts and occupational radiation doses under both normal and accident conditions for these activities.

9.1 CURRENT STATUS AND APPLICABLE CONSTRAINTS

9.1.1 Waste Management Activities to Date

Current waste management activities involve handling radioactive waste that has been produced in decontamination and other cleanup activities at TMI-2. No significant environmental impacts have been identified for the current phase of the operations. Shipment of wastes for disposal has been by truck to the commercial low-level-waste disposal facility near Richland, Washington. Wastes shipped to date, consisting of compacted waste in drums and cleanup materials in wooden boxes, have been contaminated with low levels of radioactivity. As of February 5, 1981, 2013 drums and 273 LSA boxes of low-level waste had been transferred off the island in 36 truck shipments.

An interim storage facility at the Unit 2 cooling tower desilting basin has been constructed to store wastes such as the resin bed liners from the EPICOR II system until these wastes can be processed, as necessary, and shipped for disposal. The interim storage facility presently consists of two modules, with space available for up to six modules. Each module has 60 cells, and each cell can hold two of the 4-ft × 4-ft resin bed liners or one 6-ft × 6-ft liner.

In addition, a temporary radwaste surface facility is located in this general area (see Fig. 1.3).

9.1.2 Waste Package and Shipment Parameters

In Sections 5 through 8, the volumes of each type of waste generated from the accident cleanup and decontamination operations have been estimated. In addition, best-case (minimum number of packages) and worst-case (maximum number of packages) estimates have been made of the number of packages that will be required for each waste form. In this section the totals of each type of package are provided for all the waste to be shipped from TMI-2 under best-case and worst-case conditions. The number of shipments of each type of package are then projected.

The regulatory and technical constraints that influence the shipment parameters are discussed in Section 9.1.3. Shipment of any radioactive material must comply with regulations on external dose rate. Shielding is generally not necessary to meet these regulations for truckloads of drums or boxes of low-activity waste material, but it is necessary for higher-activity wastes such as ion-exchange materials (dewatered or solidified) and damaged irradiated fuel. For purposes of estimating numbers of shipments when shielded casks or overpacks are used, the shipments will be called shielded shipments; when such shielded casks or overpacks are not used, the shipments will be called unshielded shipments. The number of unshielded packages (in the form of 55-gallon drums and low-specific-activity, LSA, boxes) and the number of unshielded waste shipments for each of these categories of waste are listed in Table 9.1 for best and worst cases. The summary in Table 9.1 was compiled from the information in Section 8. The waste in drums includes compacted trash and immobilized decontamination liquids of low specific activity. The waste packaged in LSA boxes includes noncompactible trash, contaminated equipment, and low -activity irradiated hardware.

The large quantity of packages containing EPICOR II and zeolite system liners, drums containing other high-specific-activity waste, and the fuel casks will be shielded shipments. The total number of packages of each type and the number of shielded shipments for the best- and worst-case conditions are given in Tables 9.2 through 9.5.

Table 9.1. Estimated Number of Unshielded Waste Shipments

	Best-Case Conditions		Worst-Case Conditions	
Type of Waste	Number of Packages	Number of Shipments	Number of Packages	Number of Shipments
55-Gallon Drums[a]				
Trash	6,400[b]	53	13,000[c]	108
Decontamination solutions (AFHB and reactor building)	1,600	14	2,400	20
Drum totals	8,000	67	15,400	128
LSA Boxes[d]				
Trash	1,025	86	1,790	149
Contaminated equipment and hardware	17	2	338	28
Box totals	1,042	88	2,128	177
Totals		155		305

[a] Based on surface radiation level distribution with 25 percent of drums at 50 mR/hr, 50 percent at 100 mR/hr, and 25 percent at 200 mR/hr. Average load is about 120 drums per shipment.

[b] Under best-case conditions, trash incineration will reduce number of trash drums to 1600. This will reduce number of low-level trash shipments from 53 to 13. Incineration will, however, produce an additional 480 drums requiring 34 shielded drum shipments.

[c] Under worst-case conditions, trash incineration will reduce number of trash drums from 13,000 to 3,260. This will reduce number of low-level trash shipments from 108 to 27. Incineration will, however, produce an additional 974 drums requiring 70 shielded drum shipments.

[d] Based on surface radiation level distribution with 25 percent of boxes at 50 mR/hr, 50 percent at 100 mR/hr, and 25 percent at 200 mR/hr. Average load is 12 boxes per shipment. Wooden boxes are assumed, but metal LSA boxes are a feasible alternative.

Table 9.2. Estimated Number of Shielded Ion-Exchange Material Shipments

Source of Treated Liquid Waste	Best-Case Conditions					Worst-Case Conditions				
	Prefilters	Zeolite Liners	Cation Liners	Mixed Bed Liners	Total	Prefilters	Zeolite Liners	Cation Liners	Mixed Bed Liners	Total
AFHB water	49[a]	-	14[a]	6	69	49[a]	-	14[a]	6[a]	69
Reactor building sump water[b]	-	6	1	1	8	-	27	4	2	33
RCS accident water[c]	-	1	1	1	3	-	3	7	3	13
RCS flush and drain water[c]		1	1	1	3	-	6	35	8	49
RCS decontamination solutions[d]	-	-	1	1	2	-	3	2	1	6
Total number of shipments	49	8	18	10	85	49	39	62	20	170

[a] EPICOR II system wastes currently in storage.

[b] Best case assumes zeolite liners loaded to 120,000 Ci; worst case assumes 10,000 Ci loading.

[c] Best case assumes SDS processing; worst case assumes modified EPICOR II system is used.

[d] Best case is evaporation or bituminization; worst case assumes SDS.

Table 9.3. Estimated Number of Shielded Drum Shipments

Waste Type and Drum Surface Radiation Level	Best-Case Conditions			Worst-Case Conditions		
	Number of Drums	Drums per Shipment	Number of Shipments	Number of Drums	Drums per Shipment	Number of Shipments
Sludge						
> 5 R < 20 R/hr	--	--	--	15	14	1
> 20 R < 500 R/hr	22	4	6	22	4	6
Incinerator Ash[a]						
> 1 R < 2 R/hr	480	14	34	974	14	70
Spent Filters						
> 10 R < 100 R/hr	4	7	1	29	7	5
Immobilized Evaporator Bottoms						
> 2 R < 20 R/hr	b/	--	--	1670	14	119
Totals			41			201

[a] Incinerator ash drums produced if trash incineration alternative is implemented. See footnotes b and c in Table 9.1.

[b] Evaporator bottoms are not generated under best-case conditions when the CAN DECON technique is used for primary water system decontamination.

Table 9.4. Estimated Number of Miscellaneous Shielded Shipments

Type of Waste	Best-Case Conditions		Worst-Case Conditions	
	Number of Packages	Number of Shipments	Number of Packages	Number of Shipments
Contaminated equipment	38	13	-	-
Mirror insulation	-	-	86	86
Core filter	6	6	6	6
Irradiated hardware	15	15	105	105
Zeolite system filters	17	6	33	11
Totals		40		208

Table 9.5. Estimated Number of Shielded Shipments in Fuel Casks

Type of Shipment	Best-Case Conditions		Worst-Case Conditions	
	Number of Casks	Number of Shipments	Number of Casks	Number of Shipments
Damaged fuel assemblies	50	50	177	177
Core particulates (debris)	6	6	6	6
Totals	56	56	183	183

The number of ion-exchange liner shipments under best- and worst-case conditions are summarized in Table 9.2. Ion-exchange material liners (10-ft^3 to 195-ft^3) would be shipped in licensed shipping casks with a capacity of one liner each. The choice of alternatives to process reactor building sump and primary system water discussed in Section 7.1 will have a significant impact on the number of liners to be shipped offsite.

Waste packaged in 55-gallon drums with surface radiation levels above 200 mR/hr is assumed to be transported in a shielded shipping overpack. The number of drums that could be generated under best- and worst-case conditions and the maximum surface radiation levels for these drums are given in Table 9.3. Two of the waste forms--accident sludge and spent filter cartridge assemblies--will be generated regardless of the alternatives selected. The other waste forms will be generated only if the treatment alternative which leads to their generation is implemented.

If an incinerator is used for combustible trash, the number of drums containing compacted trash shown in Table 9.1 will be reduced by a factor of 10, but between 480 and 974 drums containing immobilized ash with surface radiation levels of up to 2 R/hr will have to be shipped offsite in shielded casks.

If the alkaline permanganate solution is used to decontaminate the primary system, 1670 drums of immobilized evaporator bottoms with surface radiation levels of up to 16 R/hr could be generated. As shown in Table 9.6, the estimated minimum number of total shipments is 353 and the maximum is 997.

9.1.3 Regulatory and Technical Constraints

There are four basic safety requirements that must be met when radioactive materials are transported:

1. Adequate containment of the radioactive material.
2. Adequate control of the radiation emitted by the material.
3. Safe dissipation of heat generated in the process of absorbing the radiation.
4. Prevention of nuclear criticality, i.e., prevention of the accumulation of enough fissile material in one location to result in a nuclear chain reaction.

The transportation of radioactive materials within the United States is regulated by the Nuclear Regulatory Commission (NRC) and the Department of Transportation (DOT). Part 71 of Title 10 of the Code of Federal Regulations contains applicable NRC rules and regulations. NRC regulations provide the standards which must be met, rather than attempt to specify how they are to be met. An example of the application of this basic concept is the fact that the regulations do not prohibit the shipment of any specific radioisotope as long as the basic safety standards are met.

The Department of Transportation (DOT), under the Department of Transportation Act of 1966, the Transportation of Explosives Act, the Dangerous Cargo Act, the Federal Aviation Act of 1958, and the Transportation Safety Act of 1974, has regulatory responsibility for safety in transportation. The DOT regulations governing carriage of radioactive materials by rail and by common, contract, or private carriers by public highway (e.g., truck) are found in 49 CFR Parts 170-189. The DOT regulations governing packaging of radioactive materials, which are consistent with the NRC regulations, are found in 49 CFR Parts 173 and 178.

9.1.3.1 Packaging

Applicable Regulations

Packaging for radioactive materials is determined by the amount, kind, and physical form of the radioactive material to be transported.

In defining the packaging standards and the package content limits, the consequences of loss of containment must be considered. In the event of radioactive material release, a hazard to transport workers and to the general public exists because of the external radiation emitted from the exposed radionuclides and the often more serious problem of intake into the body, particularly through ingestion or inhalation. The radiotoxicity hazards of radionuclides vary by eight orders

Table 9.6. Summary of Estimated Number of Shipments

Type of Waste	Best-Case Conditions	Worst-Case Conditions
Low-level solids		
Drums - trash	13[a]	108
LSA boxes - trash	86	149
LSA boxes - equipment and hardware	2	28[b]
LSA boxes - mirror insulation	16[c]	-
Immobilized decontamination liquids		
Unshielded drums	14	20
Shielded drums (evap. bottoms)	None	119
Shielded ion-exchange materials		
AFHB water	69	69
Reactor building sump water	8	33
RCS accident water	3	13
RCS flush and drain water	3	49
RCS decontamination solutions	2	6
Shielded drums		
Accident sludge	6	7
Spent filters	1	5
Incinerator ash	34[a]	-
Miscellaneous shielded shipments		
Contaminated equipment	13[b]	-
Mirror insulation	-	86[c]
Core filters	6	6
Irradiated hardware	15	105
Zeolite system filters	6	11
Damaged fuel assemblies (and core debris)	56	183
Totals	353	997

[a] Best case for trash drums includes generation of 34 shielded incinerator ash drums.

[b] Contaminated equipment can be packaged in unshielded 80 ft^3 LSA boxes (worst-case conditions) or shielded 70 ft^3 liners (best-case conditions).

[c] Mirror insulation can be packaged in unshielded 80 ft^3 LSA boxes (best-case conditions) or shielded 70 ft^3 liners (worst-case conditions).

of magnitude. Standards have been developed that take into account the toxicity of each radioisotope that is being transported. For this reason, each radioisotope is classified, for transport purposes, into one of the seven transport groups, according to its relative toxicity. A list of the radionuclides and their respective transport groups is found in 10 CFR Part 71, Appendix C, and in 49 CFR Part 173.390.

Radioisotope quantity limits are established for each transport group, in order of increasing quantity, as limited quantity, Type A, Type B, and large quantity. These quantity limits then establish the use of either Type A, Type B, or low-specific activity (LSA) packaging. These categories are further defined as:

- Type A and Type B--The distinction between Type A and Type B packaging is significant. Since Type B packages carry larger quantities of radioactive materials than Type A, they are designed to more stringent standards and are considerably more accident resistant than Type A packages. Type A and Type B packages are both NRC-certified based on testing and design data submitted to the NRC.

  Type A packages, in addition to having adequate radiation shielding, are designed to withstand normal transportation conditions. Limitations on Type A package contents are such that an intake of one-millionth of the maximum allowable package contents would not result in a radiation dose to any organ of the body exceeding internationally accepted limits nor a radiation dose greater than 1 rem/hr at 10 ft from the unshielded contents.

  Type B packages often must be heavily shielded and are designed, based on testing and engineering analysis, to withstand severe accident conditions as well as normal conditions. These packages would be expected to survive a severe accident without any significant release of their contents. In spite of the demonstrated integrity of Type B packages, to be conservative, some of the accident scenarios considered in this statement include releases from Type B packages.

- LSA Materials--Frequently, large volume shipments (e.g., compacted trash) are transported in Type A packages. Strong, tight industrial packaging is permitted for Type A quantities of LSA materials moved in exclusive use vehicles.

Available Packages

Shipments of wastes requiring shielded Type A or Type B packages will be made using available certified shielded casks. Waste containers have been designed to fit into these existing casks. The limited number of available casks, however, may constrain waste shipment schedules. The purchase or lease of additional casks, which is being considered, would alleviate this situation.

9.1.3.2 Transportation

Applicable Regulations

Adequate control of the radiation is required when transporting radioactive material. To meet the radiation control limits, the shipper must often provide necessary shielding as an integral component of the packaging of the material.

Because TMI-2 waste and fuel shipments will be consigned for sole use, the following dose limits specified in 49 CFR Part 173.393(j) apply:

1. 1,000 mrem/hr at 3 ft from the external surface of the package (closed transport vehicle only);
2. 200 mrem/hr at any point on the external surface of the car or vehicle (closed transport vehicle only);
3. 10 mrem/hr at any point 6 ft from the vehicle planes projected by the outer lateral surface of the car or vehicle; or if the load is transported in an open transport vehicle, at any point 6 ft from the vertical planes projected from the outer edges of the vehicle.

4. 2 mrem/hr in any normally occupied position in the car or vehicle, except that this provision does not apply to private motor carriers.

Based on these constraints, certain portions of the TMI-2 waste material can be shipped in unshielded packages; others will require shielded packages. Vehicles carrying the waste packages will be placarded and marked in conformance with DOT requirements.

Shipping Configurations

The shipping configurations used to transport packaged radioactive waste from TMI-2 to another treatment, storage, or disposal facility depend on the types of radionuclides in the waste, their gross activities, the type of disposal container and its radiation level. All shipments must conform to radioactive material regulations, as well as transportation regulations.

To date, radioactive waste shipments from TMI-2, with a few special exceptions, have been limited to legal-weight truck transport. Most states require that the total weight of the tractor, trailer, and package be less than about 73,000 pounds. Within this weight constraint, two categories of shipments (i.e., shielded and unshielded) were considered for low-level radioactive waste.

Most containers for low-activity waste are metal drums or wooden boxes. Materials such as dewatered ion-exchange materials or damaged fuel may contain considerable radioactivity. Shipment of any radioactive material must comply with regulations on external dose rate. Shielding is generally not necessary to meet these regulations for truckloads of drums or boxes of low-activity waste material, but is necessary for high-specific-activity wastes such as ion-exchange resins (dewatered or solidified) and irradiated fuel. In certain cases, aggregates of drums or boxes of low-level waste must be shielded, and they may be loaded into shielded overpacks for shipment. The shielded casks or overpacks supply additional mechanical integrity to the truck contents, for prevention of uncontrolled releases to the environment, in both normal transportation and in transportation accidents.

The legal weight constraint limits the total payload (the amount of packaged waste) plus shielding (if required) to about 42,000 pounds. The shipment payload is further limited by the shipping cask internal dimensions and its compatibility with the type of disposable container used. The characteristics of available containers for the TMI-2 waste are presented in Table 9.7.

The shipping configurations considered for unshielded and shielded TMI-2 packaged waste are shown in Tables 9.8 and 9.9 as a function of the containers shown in Table 9.7, the radiation level at the surface of the disposable container, and available shipping casks.

Table 9.7. Characteristics of Disposable Containers

Type of Container	Approximate Dimensions[a] (ft) Diameter × Height	Internal Volume (ft^3)	Waste Capacity (ft^3)
55-gallon drum	2 × 3	7.35	7
10-ft^3 liner	2 × 4.5	10	8-10
50-ft^3 liner	4 × 4	44	40
75-ft^3 liner	4.5 × 5.5	75	70
80-ft^3 liner	5.5 × 5.5	80	75
180-ft^3 liner	6 × 6	175	165
195-ft^3 liner	6.5 × 6.5	195	180
LSA box[b]	[c]	81	80

[a] D = diameter; H = height.

[b] Unshielded configurations only, all others are compatible with shielded shipment.

[c] LSA box is 3 ft × 4 ft × 6.5 ft.

Table 9.8. Unshielded Shipping Configurations

Radiation Level At Container Surface	Shipment Waste Capacity (ft^3)			
	55-Gallon Drums		LSA Boxes	
	Volume	Number	Volume	Number
50 mR/hr	840	120	960	12
50 to 100 mR/hr	630	90	720	9
100 to 200 mR/hr	350	50	480	6

Table 9.9. Shielded Shipping Configurations

Radiation Level At Container Surface	Shipment Waste Capacity (ft^3)[a]	
	55-Gallon Drums[b]	Large Containers
To 1 R/hr	130	200
To 10 R/hr	98	170
To 20 R/hr	98	170
To 100 R/hr	49	40 to 75[c]
To 1,000 R/hr	28	40 to 75[c]
To 10,000 R/hr	14	20

[a] Partial list of containers used.

[b] Divide by 7 to obtain number of drums.

[c] Two different-sized disposable containers.

Shipping Methods

Shipments by truck and intermodal rail and truck are alternatives for the transport of TMI wastes. Final selection may depend on the choice of the disposal site. Trucks can depart from TMI and go directly to any of the potential storage or disposal locations. While railroad facilities are available on the TMI site, there are no rail facilities at a number of the potential storage and disposal sites. For example, the nearest railroad to the Beatty, Nevada, disposal site is about 115 miles away in Las Vegas, Nevada. There is a rail siding on the Hanford site about one mile from the commercial disposal site. Unless this siding is extended to the disposal site, a transfer by truck would be necessary. Special arrangements for unloading rail cars using Hanford site facilities would have to be made with DOE.

In addition, availability of shielded rail casks is limited; there are constraints imposed on rail shipments that create logistics problems and increase shipment duration; and intermodal shipments may involve higher exposure levels to handlers than for single mode shipments because of the transfers required.

Therefore, truck shipment from TMI to the treatment, storage, or disposal locations is considered to be the most likely mode of shipment for the majority of the TMI-2 waste. The projected transport routes are discussed in Section 3.2, and the impact analyses in Section 9.4 are based on the use of truck shipments for the waste and damaged fuel.

Routing Considerations

Until recently, the primary safety measures in regulation of radioactive materials transportation were controls on packaging and related transportation parameters. However, the DOT has now enacted regulations (January 19, 1981; effective February 1, 1982) that focus on routing and related operational controls for highway transportation of certain radioactive materials, including waste. These regulations are revisions to 49 CFR Parts 173 and 177. NRC approval of routing for spent fuel shipments is also now required (10 CFR Section 73.37).

In brief, the regulations require the following:

- A general rule would require a motor vehicle carrying radioactive material that is placarded to be operated on a route that presents a risk to the fewest persons, unless there is not any practicable alternative highway route or it is operated on a "preferred" highway. The motor vehicle would thus have to be operated on a route which minimizes transit times, so as to minimize exposure.

- A more specific rule would require that any motor vehicle transporting a package containing a "large quantity" of radioactive materials be operated on "preferred" highways in accordance with a written route plan prepared by the carrier before departure. Preferred highways would be designated by state agencies based on a policy of an overall minimization of impacts from normal transportation and from transportation accidents. This rule would require use of an interstate urban circumferential, or bypass, route to avoid cities if available, instead of an interstate through route.

- Notification to states of Type B shipments.

9.1.3.3 Treatment (Offsite), Storage, and/or Disposal

Availability of Facilities

The potential alternatives for management of TMI waste and fuel include use of offsite treatment, storage, and disposal facilities. The use of these facilities is dependent on the characteristics of the waste, the costs and benefits of each alternative, and the capabilities and availability of the facilities for their handling. Other factors, as discussed in the following sections also need to be considered regarding disposition of these waste forms.

Low-Level Waste (LLW)

The only offsite alternative considered for LLW is burial in a shallow land burial site. Consistent with current LLW management practices, no offsite treatment or storage alternatives are evaluated for LLW management.

Only three of the six commercial LLW burial sites currently are operative. These three are the sites at Beatty, Nevada; Richland, Washington; and Barnwell, South Carolina. Of these three, the Barnwell facility is currently excluded from receiving TMI-2 accident cleanup wastes by order of the Governor, and without modification of that exclusion leaves the Beatty and Hanford facilities as the only potential sites currently available for receipt of LLW from TMI-2. As a result of an initiative passed in the State of Washington in November 1980, the Richland facility will not be permitted to accept out-of-state shipments of reactor plant wastes after July 1981 unless a specific "compact" is negotiated on an individual case basis. The governors of South Carolina and Washington have urged the development of regional LLW disposal sites in other parts of the country to reduce the need for continuing long-range shipments of high volumes of wastes to the sites in their states.

In December 1980, the Low-Level Radioactive Waste Policy Act was enacted (Congressional Record, 516545, December 13, 1980). This act stated that each state is responsible for providing for the availability of disposal capacity for commercial low-level wastes generated within its borders. The act also allowed states to enter into compacts necessary to provide for the establishment and operation of regional disposal facilities. These compacts would require Congressional consent prior to taking effect and could restrict the use of the regional facilities to states within the region after January 1, 1986, but not before that time.

While the Low-Level Radioactive Waste Policy Act assigns responsibility to each state for providing low-level radioactive waste disposal capacity, the federal government recognizes a need to assist states in dealing with unique wastes, such as those wastes which will result from the TMI-2 cleanup, which may not be suitable for normal shallow land disposal. The act, however, emphasizes a need for state authorities to site new disposal facilities or enter into agreements needed to provide for the adequate disposal capacity for the low-level wastes generated within that state.

Congress also has requested that the DOE, as the lead federal agency for nuclear waste management, prepare a preliminary assessment pertaining to the development of regionally distributed disposal sites for LLW. The DOE has estimated that [illegible] to eight additional sites will be needed within the next ten years and that the Northeast and Midwest sectors of the country are the regions with the greatest near-term need for disposal facilities. The establishment of a new LLW burial site in the state of Pennsylvania to accommodate TMI waste also is an option. Such a facility would provide a minimal transport distance for the waste shipped from TMI, and also would permit disposal to be accomplished in the state where the waste was generated. However, no new land burial sites are in the development stage, and it is likely that five to ten years will be required before new sites are available. Thus, early direct shipment to a new LLW disposal site is not a near-term option.

Another option for the disposal of the TMI-2 LLW is to reopen existing commercial sites currently not receiving waste. There are three sites in this category--West Valley in New York; Maxey Flats in Kentucky; and Sheffield in Illinois. The Sheffield site has limited capacity, but the capacity at the other two sites would be sufficient for the TMI-2 LLW. The Maxey Flats site currently is constrained from receiving any waste shipments by state legislative order, and plans are being developed for decommissioning of this facility. The West Valley site is the closest LLW burial facility to TMI. It would, thus, appear to be a technically feasible alternative. However, the future disposition of this facility, as well as other waste management facilities at West Valley, cannot now be projected. Recent Congressional legislation (West Valley Demonstration Project Act-Public Law 96-368, dated October 1, 1980) has committed the DOE to participate with New York State in a program to solidify and dispose of the HLW stored at the site, but the legislation has not resolved the future utilization of the burial locations.

A number of the federally (DOE) owned LLW disposal facilities also are alternative backup locations for disposal of TMI-2 LLW waste. The DOE facilities at Hanford, Washington; the Savannah River Plant in South Carolina; and the Idaho National Engineering Laboratory are suitable locations with sufficient capacity for additional LLW burial. Current DOE policy does not allow for this option, and the DOE has not indicated that any exception would be made for TMI-2 waste at this time.

Interim onsite storage of LLW packages prior to shipment is anticipated, and facilities are now under construction or in use for temporary storage of certain of the packages.

High-Specific-Activity Waste

In the case of the management of the high-specific-activity waste (HSAW), offsite alternatives include the use of facilities for special packaging or treating the waste, interim or long-term storage, and disposal in a range of potential facilities, bounded by disposal in a geologic repository at one extreme and intermediate depth burial at the other.

The capability of both commercial and governmental organizations to undertake the special handling, processing, long-term storage and disposal of the HSA first-stage ion-exchange wastes has been considered. There are several commercial organizations in the fuel fabrication field that have basic hot-cell units that have been used for spent fuel examination, and have the equipment and staff for handling these types of radioactive waste materials. However, none of these commercial organizations has routinely handled the immobilizaton, long-term storage, or disposal of such large quantities (thousands of curies) of long-lived (30-year half-life) radionuclides, nor do they have experience or facilities for processing such wastes to achieve a form suitable for long-term storage or disposal. On the other hand, there are commercial firms in the waste handling field that have experience in handling and immobilization of essentially normal reactor wastes that do not have the special facilities and capabilities to handle the HSA first-stage wastes.

In overview, while some of the firms might have facilities and staff along with some interest in developing processes for such work, and others might have business interests in actual waste immobilization, none has the required combination of extended availability of special facilities, specialized staff, and waste management business interest that would have to be applied to carry out such a waste management task. Thus, due to its special nature and limited application, adequate commercial capability is not available (and commercial interest to develop that capability is not evident) for the handling of high-specific-activity special TMI-2 wastes.

On the other hand, the DOE, through its contractors and laboratories, as a result of efforts to develop suitable methods for immobilization and disposal of a wide variety of government and commercial radioactive wastes, appears to have the only suitable combination of established personnel, technological capabilities, and interests for developing and carrying out such special operations for a one-time application. Accordingly, in analyzing the alternatives for the handling, processing, long-term storage or disposal of high-specific-activity TMI-2 wastes, consideration of generally capable DOE sites or facilities has been included.

Treatment alternatives include high-integrity packaging, processing and immobilization, incineration, chemical digestion, and elution and immobilization of the eluate as described in Section 8.1. The only currently available offsite treatment facilities for special processing are located at federally owned (DOE) sites, where the TMI waste could be either treated separately or possibly comingled with the DOE and DOD waste. However, these facilities currently have not been made available for treatment of TMI-2 waste because the DOE is concerned that NRC's regulatory authority would be extended to the entire DOE facility. The DOE has acknowledged[1] that only its facilities have the unique technical capabilities (both staff and equipment) to handle these wastes and that nontechnical considerations have dictated their policies on this question to date. Requests have been made by the NRC to the DOE[2] to accept these waste materials, and clarification provided of the NRC's position that no licensing would be required if the DOE facility's primary functions were other than handling commercial wastes in accordance with the Energy Reorganization Act, Section 202.[3] The DOE sites where these treatment processes could be conducted, if permitted, are as follows:

- Processing and Immobilization. Zeolites and resins could be immobilized and packaged for eventual repository disposal. DOE solidification facilities exist at the waste calcining plant at the Idaho National Engineering Laboratory and are planned for construction at the Savannah River plant in the late 1980s. A commercial pilot-plant high-level waste processing operation also is located on the nonmilitary portion of the Hanford Reservation.

- Incineration. Effective incineration of the combustible component of the HSAW (resins) to reduce volume and mass could be possible at a number of facilities being developed for treating higher specific activity waste. These facilities include a controlled air incinerator (CAI) at the Los Alamos Scientific Laboratory used for transuranic waste; a CAI unit being developed at the Savannah River Plant for operation in 1981; a cyclone incinerator being adapted for TRU waste burning at the Mound Laboratory; a rotary kiln production unit at the Rocky Flats Plant scheduled for operation in 1981; and a slagging pyrolysis unit proposed for operation in 1986 to treat transuranic waste at the Idaho National Engineering Laboratory. In order for these incineration facilities to be used for TMI-2 HSAW, systems modifications would be required.

  Acid Digestion. Acid digestion of combustible organic-based waste is accomplished at the Radioactive Acid Digestion Test Unit (RADTU) at the Hanford Engineering Development Laboratory (HEDL). The unit has been used for low-level waste and is being modified to increase its capacity. In addition, other modifications might be required to permit processing of the HSAW from TMI-2.

- Elution and Immobilization. Elution of resins and zeolites to obtain the separated fission products has been developed at the Savannah River Plant and the Hanford facilities of the DOE. The work has been done in processing waste streams currently at these facilities.

Either short-term or long-term storage of the HSAW could be carried out at the TMI site or an offsite location. However, further processing at TMI to put these materials in a form suitable for eventual disposal in a future repository is not, in the staff's opinion, an appropriate activity to be performed at TMI since the licensee's staff and industry in general have no experience in this area. On the other hand, the existence of DOE facilities, with experienced personnel, argues against construction of duplicate equipment at TMI, for a one-time application, with the attendant increase in contaminated equipment and risk to workers and the public. Onsite storage at TMI will require construction of a facility designed to accommodate the required packaging and handling equipment. Handling and space constraints attendant to use of a range of packaging configurations, and the package surface radiation levels of the HSAW, will have to be taken into account.

Offsite short-term storage at DOE storage facilities, at one of the aforementioned treatment facilities (which would minimize transportation and handling) or at a final disposal site (pending its completion), is a potential alternative for the HSAW.

The potential SDS first-stage wastes will be very special in nature and should most logically be handled like spent fuel materials. Accordingly, long-term storage options for the SDS first-stage wastes include nuclear power plant spent fuel pools if excess capacity is available, DOE facility spent fuel pools or other suitable areas; storage at the shutdown commercial reprocessing facilities, or an away-from-reactor (AFR) storage pool facility. However, AFR facilities do not currently exist and future locations have not been designated.

Under current policy, DOE storage facilities would not be available for TMI waste. Other institutional constraints on the use of specific storage facilities include the commitment on the part of the Federal Government to the State of Idaho to remove existing transuranic waste stored at the site (which would likely mitigate against any additional HSAW being stored there); the fact that the DOE facilities at the Savannah River Plant and Hanford are in states (South Carolina and Washington) which have commercial LLW disposal sites, and any constraints imposed by these states on shipping might also apply to waste shipments to a DOE facility in the same state.

The alternatives for disposal of the HSAW range from disposal in a geologic repository when one is available, which will require an extended storage period in a stable form (e.g., vitrified), to the possible use of intermediate depth burial incorporating an intruder barrier to prevent violation of the cover integrity for some of the lower end of the HSAW.

Although a national program is in progress to identify and qualify sites for geologic disposal facilities in a variety of media, specific locations have not yet been identified. Intermediate depth burial as an alternative to shallow land burial is currently being developed. This technique may be a feasible alternative for some of the HSAW. The DOE plans to demonstrate intermediate-depth burial. This demonstration could possibly use some of the HSAW, such as the EPICOR-II first-stage waste.

Fuel

Two independent assessments[4,5] have been made on the condition of the TMI-2 core resulting from the accident. Although most probable conclusions reached in the two assessments differ as to the extent of the projected damage, both conclude that the temperatures and pressures experienced by the fuel assemblies during the accident have significantly altered the fuel components to the extent that they no longer meet specifications for reuse. Lacking specific observational information, it must be presumed for planning purposes that the core will include intact fuel assemblies that may have undergone some damage and distortion, fused formations of fuel larger in cross section than an individual assembly, and loose debris.

As previously discussed, all the fuel will be retrieved from the reactor. Some portion all of the core will be sent to hot cell areas at a designated national laboratory for diagnostic evaluation. Upon completion of the diagnostic evaluation, the residue of the examined fuel could be either added to the waste stream at the laboratory for processing as high-level waste, or treated as described below for the remainder of the fuel. The remainder of the fuel would be packaged for interim storage pending final disposition. The alternatives available and the constraints affecting their availability are as follows:

- Wet (Pool) Storage - Damaged fuel can be stored either onsite in the TMI storage pool, offsite at another reactor pool, or at a regional AFR storage pool; in pool spaces at the national laboratories; or at the storage pools at the shutdown commercial reprocessing facilities. As previously noted, AFR facilities may not be available until 1983-1984 at the earliest; excess capacity at other reactor locations is extremely limited; and DOE facilities are not available at this time for commercial use (see, however, discussion provided above). In addition, this alternative represents a temporary solution for the fuel disposition, not a final one.

- Dry Storage - Another potential interim alternative involves placing the canned TMI-2 fuel in either a storage vault or hot cell, or in a specially designed spent fuel caisson. The constraints here are, again, the current unavailability of DOE or other facilities of this nature, and the temporary nature of this option pending final disposition.

- Chemical Processing at a Commercial or DOE Facility - This alternative involves the reprocessing of the fuel to recover the useful uranium and plutonium and subsequent immobilization and disposal of the high-level waste. However, current national policy precludes the implementation of the reprocessing alternative and, even if this policy were to be changed, commercial reprocessing would probably not be available until 1990 at the earliest.

For each of the above-described alternatives, final disposal will be in a geological repository which will be sited, constructed, and operated by the DOE.

9.2 ALTERNATIVE METHODS CONSIDERED

The alternatives considered for storage, transportation, and disposal of the TMI-2 waste and fuel packages are discussed in this section.

9.2.1 Storage

9.2.1.1 Onsite

Onsite storage of the waste for an interim period prior to shipment is a feasible and necessary option. Interim storage facilities are now under construction or in use for temporary storage of certain waste types (see Fig. 9.1). Interim storage with suitable monitoring for continued waste containment integrity permits timely removal of wastes from shipment staging areas, allows for decay of radionuclides with short half-lives, permits continuing cleanup operations while ultimate disposal strategies are formulated, and provides a buffer for contingencies beyond control of the licensee, such as shipping strikes, embargoes on certain waste types, or unavailability of transport equipment (casks), offsite storage, or disposal facilities.

Licensee conceptual plans now exist for construction of an interim storage and staging area for low-level radioactive waste including certain of the high-specific-activity wastes. The facility will be sized to accommodate about 800 55-gallon drums, 150 wooden boxes, each 4 ft × 3 ft × 6½ ft, and sixty 50-ft^3 evaporator bottom liners. Radiation levels from these waste packages will range up to about 500 mR/hr. Maximum radiation levels at the fence surrounding the storage facility will be less than 0.6 mR/hr. As a buffer against shipping delays or other problems, this facility will accommodate the waste expected to be generated during a six-month period.

As an alternative to the interim storage and staging area described above, a longer term storage facility for the various forms of LLW and HSAW could be considered. Based on current projections of total waste volume to be generated in decontamination activities, the facilities would have to be considerably larger than needed for the six-month storage area previously described. Evaluation of the feasibility of this facility would include consideration of the potential for failure of container integrity during longer term storage. In addition, the means for adequately handling, shipping, and disposing or storing some of the involved waste types are available offsite.

Another potential interim waste storage alternative is to place the packages in a designated area in the reactor building. While providing a readily available shielded storage facility of significant capacity, this option has a number of drawbacks. It would necessitate storing the waste packages in the reactor building while both cleanup and fuel removal operations were being conducted, requiring revisions to existing plans, potential delays in completion of these operations, and additional costs. In addition, package handling capabilities are limited at the lower levels of the building.

Alternatives for handling the damaged irradiated fuel removed from the core would be to adapt the existing pool for long-term storage or to construct new onsite facilities for long-term storage. These options will not be necessary in the time frame being considered if long-term storage facilities become available offsite and if research and development plans being formulated for the core components are carried out for the entire core.

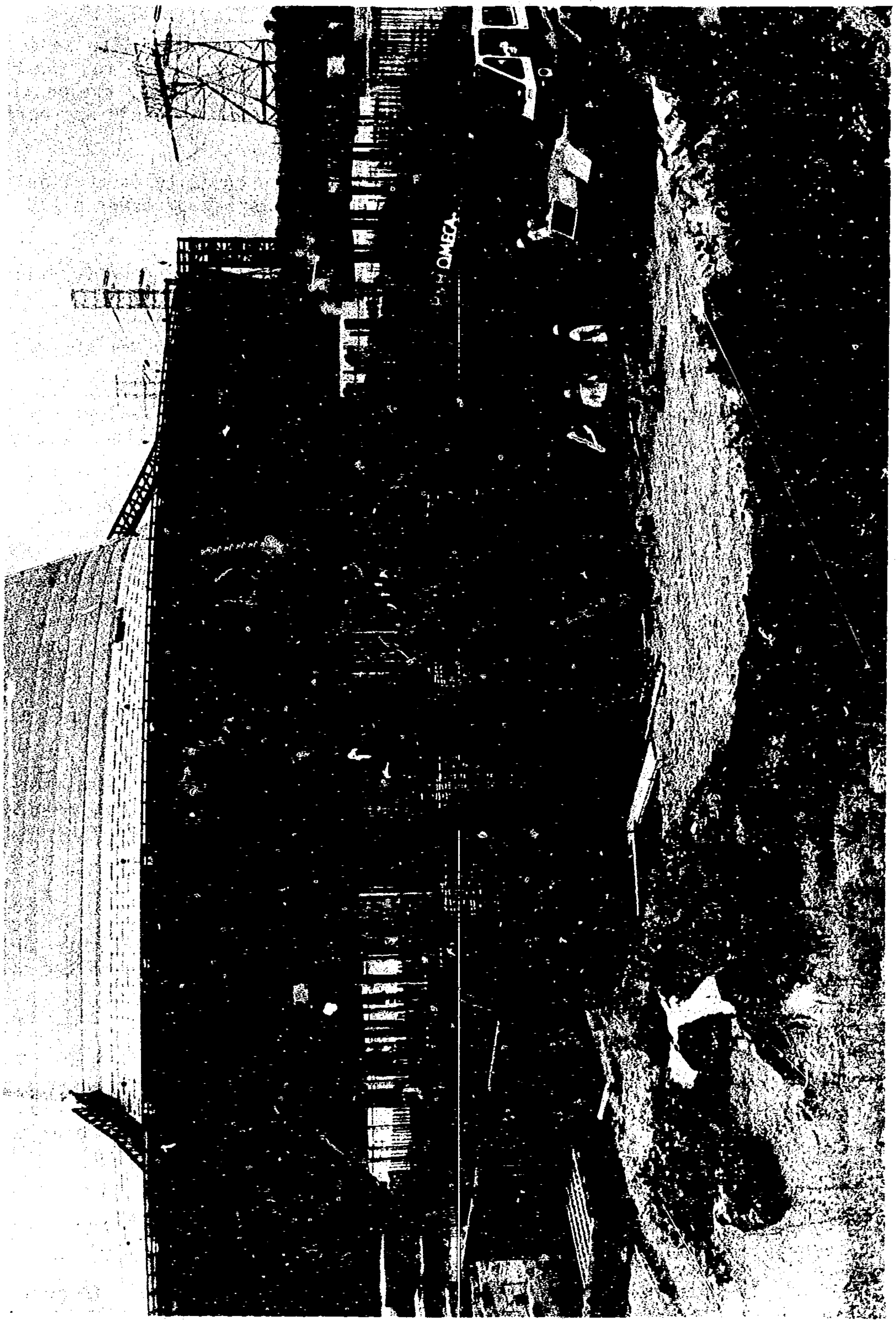

Figure 9.1. Construction of Interim Radwaste Storage Facility at TMI. Two storage modules are now completed, with individual silos surrounded by reinforced concrete. (Official TMI Photo)

9.2.1.2 Offsite

Offsite storage of the waste for periods pending a decision as to final disposal is a feasible option for the HSAW that will be disposed of by other than routine shallow land burial techniques. Offsite storage may be utilized either before or after immobilization of the waste. Storage at the same location where processing to immobilize the wastes is carried out will minimize exposure, due to reduced handling and transportation.

Offsite storage at designated storage sites, at a treatment facility, or at a site selected because of its proximity to a final disposal facility is a potential alternative for the HSAW. The following are those facility categories and specific locations that currently are considered to be prime candidates for retrievable storage facilities for TMI-2 waste because appropriate storage capacity is available or is contemplated:

- Commercial nuclear power plant spent fuel pools where excess capacity is available. It may be difficult to identify such facilities since the majority of power plants are facing a storage capacity problem.
- An AFR pool facility. As previously noted, AFR facilities do not currently exist and future locations have not been designated at this time.
- DOE-operated storage facilities, including those at the Hanford Reservation, Savannah River Plant, or Idaho National Engineering Laboratory.
- The storage pools at the commercial reprocessing facilities at West Valley, New York; and Morris, Illinois. In addition, the Barnwell, South Carolina, facility is a potential site, but has not been licensed to store nuclear material.

Long-term storage of that portion of the fuel not sent to DOE laboratories for diagnostic evaluation is a likely alternative until it is determined whether the fuel will be placed in a repository without first undergoing reprocessing or whether the fuel will be reprocessed and the HLW produced, then immobilized for disposal. The storage alternatives can be categorized as either wet storage or dry storage, as follows:

- Wet (Pool) Storage -- The available alternative locations for storage of the fuel cannisters are the same as for the HSAW.
- Dry Storage -- Involves placing the canned TMI-2 fuel in a storage vault, hot cell, or specially designed spent fuel caisson.

The duration of storage of the fuel and HSAW for final disposition in a geologic repository will depend on the program to select and to qualify a repository, which currently is not expected to be accomplished until the 1990s. Under present laws these facilities will be federally owned and NRC licensed.

9.2.2 Disposal

Different alternatives are available for final disposal of the low-level waste and high-specific-activity waste generated at TMI-2. The regulatory and institutional constraints on the availability of disposal sites are discussed in Section 9.1.3.3.

..2.2.1 Low-Level-Waste Disposal Alternatives

The feasible low-level-waste disposal alternative for TMI-2 waste generated in the near term is limited to burial in a shallow land burial site (at a maximum depth of about 30 ft below the trench cover surface) under present practice for routine wastes, the current technologically acceptable disposal approach. The potential burial sites for wastes that are similar to routinely generated wastes are as follows:

- Operating commercial LLW burial sites - The sites at Beatty, Nevada, Richland, Washington; and Barnwell, South Carolina.
- A new regional LLW disposal site in the Northeast or in Pennsylvania. (A period of interim storage of the LLW packages would be required until such a site is developed.)

- Currently shutdown commercial burial sites having available capacity--the sites at West Valley in New York and at Maxey Flats in Kentucky.
- DOE-owned LLW burial sites having available capacity--among the major DOE LLW disposal sites, the facilities at Hanford, Washington; the Savannah River Plant in South Carolina; and the Idaho National Engineering Laboratory could be suitable locations.

9.2.2.2 Alternatives for Disposal of High-Specific-Activity Waste

The alternatives for disposal of the HSAW range from disposal in a geologic repository when one is available (which will require that the waste be stored for an extended period in a stable form) to the use of intermediate depth burial incorporating an intruder barrier (i.e., layer of artificial material) to prevent violation of the cover integrity, depending on the radionuclide inventory and specific activity involved.

In the case of disposal in a geologic repository, although a national program is in progress to identify and qualify such facilities in a variety of media, specific locations have not yet been identified.

Intermediate depth burial (at a level 50 to 60 ft below the trench cover surface) represents an alternative for disposal of the lower range of the HSAW because it provides significantly better isolation from intrusion and disturbance from natural disasters (e.g., flooding) than shallow-level burial at sites where hydrogeology ensures at least comparable isolation from groundwater. A number of the existing commercial and DOE shallow land burial sites provide suitable hydrogeology for intermediate depth burial. Among the potential suitable alternatives are:

- The commercial burial sites at Richland, Washington, and Beatty, Nevada, would permit intermediate depth burial with a remaining distance in excess of 200 ft to the nearest aquifer.
- The currently shutdown burial site at West Valley, New York. The "hulls" burial area at this site has been previously used for burial of TRU and other HSAW at depths comparable to intermediate depth burial.
- The DOE burial sites at Hanford, Washington; Idaho National Engineering Laboratory; and Los Alamos are suitable for intermediate depth burial because of their hydrogeologic characteristics.

9.2.2.3 Fuel and High-Level Waste

Final disposal of the fuel and the high-level waste generated from reprocessing the fuel (if implemented) will be in a geologic repository which will be sited, constructed, and operated by the DOE.

9.2.3 Transportation

9.2.3.1 Transportation Mode

Truck shipment and combination of truck and rail are potential options for shipment of the TMI-2 waste and spent fuel packages. TMI-2 wastes shipped to date have been sent exclusively by truck.

When waste packages are shipped primarily by unshielded vehicle directly to a commercial shallow land burial site, truck transport is the preferred mode. There are no rail facilities at the likely disposal sites, and intermodal rail and truck shipment would be required if rail shipment from TMI is used. Truck shipment also would be the likely choice for shipment of the majority of the intermediate-level waste for either immobilization at a treatment facility or storage pending final disposal. In this instance, shielded vehicle shipments may be required.

Rail transport may be preferable in those cases when shielding needs for certain intermediate-level waste packages and for irradiated fuel cannisters necessitate the use of large, heavy casks and when off-loading rail spurs are available near the storage or disposal location.

Another factor that may influence the selection of rail or truck transport is the availability of the appropriate types of shielded casks at the time shipments are being scheduled.

9.2.3.2 Transportation Routes to Potential Treatment, Storage, and Disposal Facilities

The truck transportation route to the alternative treatment, storage, and disposal facilities that have been identified are described in this section. These routes, which are the currently established "most likely" routes used by nuclear waste shippers, make use of the federal interstate highway system except for short distances near the starting and termination points, where local and state roads are used. They are the routes that the waste shippers use with the greatest frequency, but are not necessarily the only routes traversed. The final selection of the routes at the time of shipment will involve consideration of the requirements of applicable DOT regulations (Sec. 9.1) on routing and specific constraints imposed by the states and municipalities through which the shipments will pass.

Routes to Treatment Facilities

The potential locations for treatment of HSAW have been identified as the DOE facilities at the Hanford Reservation, the Savannah River Plant, and the Idaho National Engineering Laboratory. Local routing of truck shipments from TMI to the interstate highway system, which is applicable for shipments to all potential locations, is shown in Figure 9.2. The complete interstate routing to Hanford, Savannah River, and INEL is shown in Figure 9.3. Local routing at each of these locations is shown in Figures 9.4, 9.5, and 9.6, respectively.

Routes to Offsite Storage Facilities

The specific locations identified in Section 9.2.1.2 as potential interim storage facilities for the HSAW or fuel include the three DOE facilities discussed above as possible treatment facilities. The routing to the Hanford Reservation, Savannah River Plant, and Idaho National Engineering Laboratory are as shown in Figures 9.3 (interstate routing), 9.4, 9.5, and 9.6, respectively (local routing).

In addition, the routes to the pool storage at the reprocessing facilities at West Valley, New York; Barnwell, South Carolina; and Morris, Illinois, which are also possible storage locations, are shown in Figure 9.7.

Routes to Disposal Facilities

For the low level waste, the routes to the commercial sites at Richland, Washington; Beatty, Nevada; and Barnwell, South Carolina, are shown in Figure 9.8. The routes to the shutdown commercial burial sites at West Valley, New York, and Maxey Flats, Kentucky, also are shown on this map. Figures 9.9 through 9.12, respectively, show the local routing at Beatty, Barnwell, West Valley, and Maxey Flats. The routes to the DOE burial sites at Hanford, Savannah River, and INEL are the same as shown in Figures 9.3 through 9.6.

Intermediate depth burial of some of the lower range of HSAW can be considered at the commercial burial sites at Richland and Beatty; the shutdown site at West Valley; and the DOE sites at Hanford, INEL, and Los Alamos. The routes to these facilities, with the exception of Los Alamos, are as shown previously. The interstate routing to Los Alamos is shown in Figure 9.3, and the local routing in Figure 9.13.

9.3 DETAILS OF METHODS AND FACILITIES

9.3.1 Storage

The onsite storage facilities currently in use or under construction are needed as an interim storage and staging area prior to shipment of wastes. The use of offsite storage for certain of the HSAW or the fuel, while technically and operationally feasible, depends on decisions yet to be made as to final disposition of the material. The availability of specific government facilities also needs to be determined (see discussion in Sec. 9.1.3.3).

The onsite concrete storage facilities for interim storage of certain of the HSAW (e.g., EPICOR II spent liners) are described in Appendix Q.

All onsite package-handling and loading operations would be conducted in a manner that ensures that occupational radiation doses are kept as low as reasonably achievable. Remote operations will be employed and shielding will be used to minimize the radiation fields to which handlers are exposed.

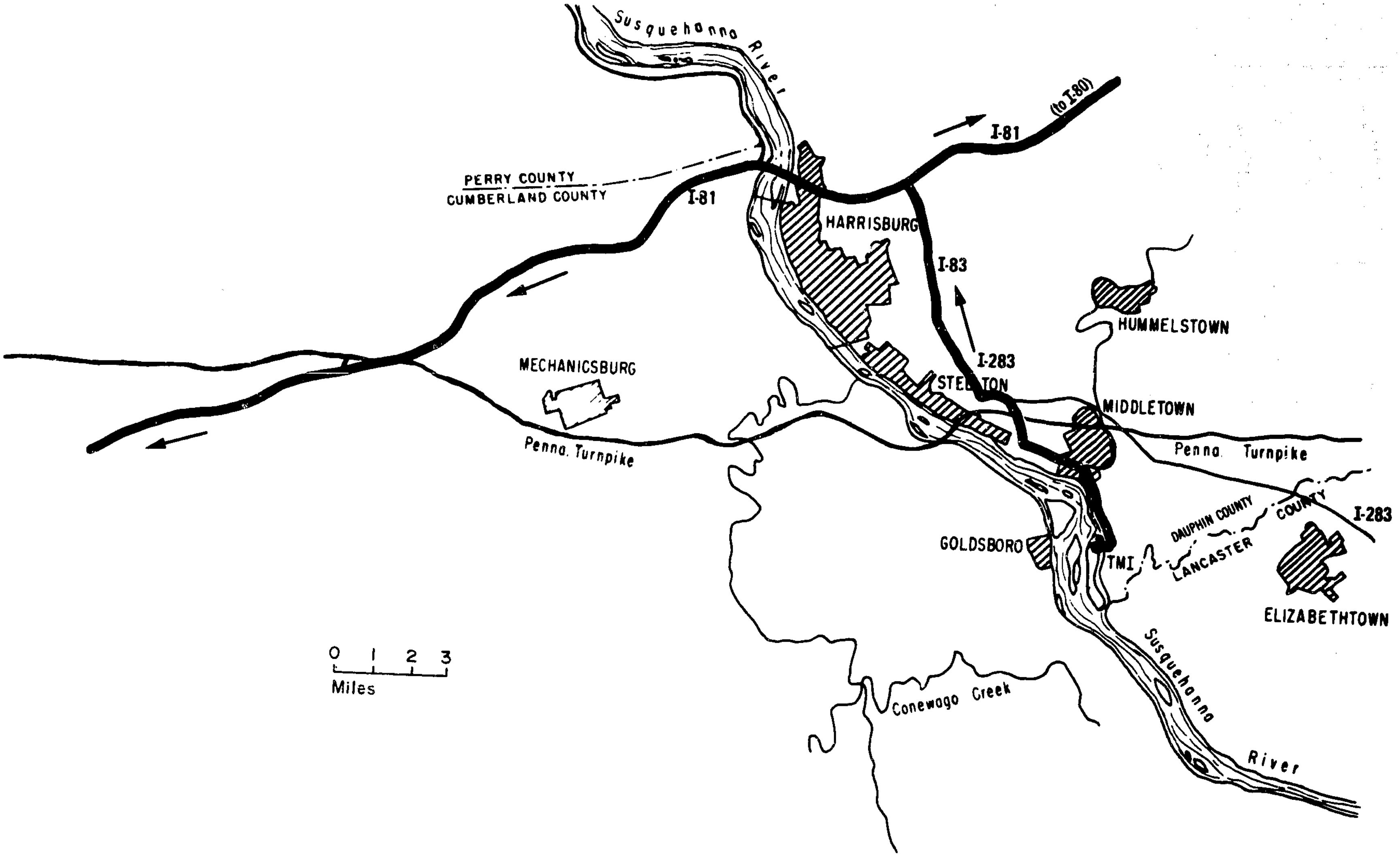
Susquehanna River
I-81
(to I-80)
PERRY COUNTY
CUMBERLAND COUNTY
I-81
HARRISBURG
I-83
HUMMELSTOWN
I-283
STEELTON
MIDDLETOWN
MECHANICSBURG
Penna. Turnpike
Penna. Turnpike
DAUPHIN COUNTY
LANCASTER COUNTY
I-283
GOLDSBORO
TMI
ELIZABETHTOWN
0 1 2 3
Miles
Conewago Creek
Susquehanna River

Figure 9.2. Route from TMI through Harrisburg Area.

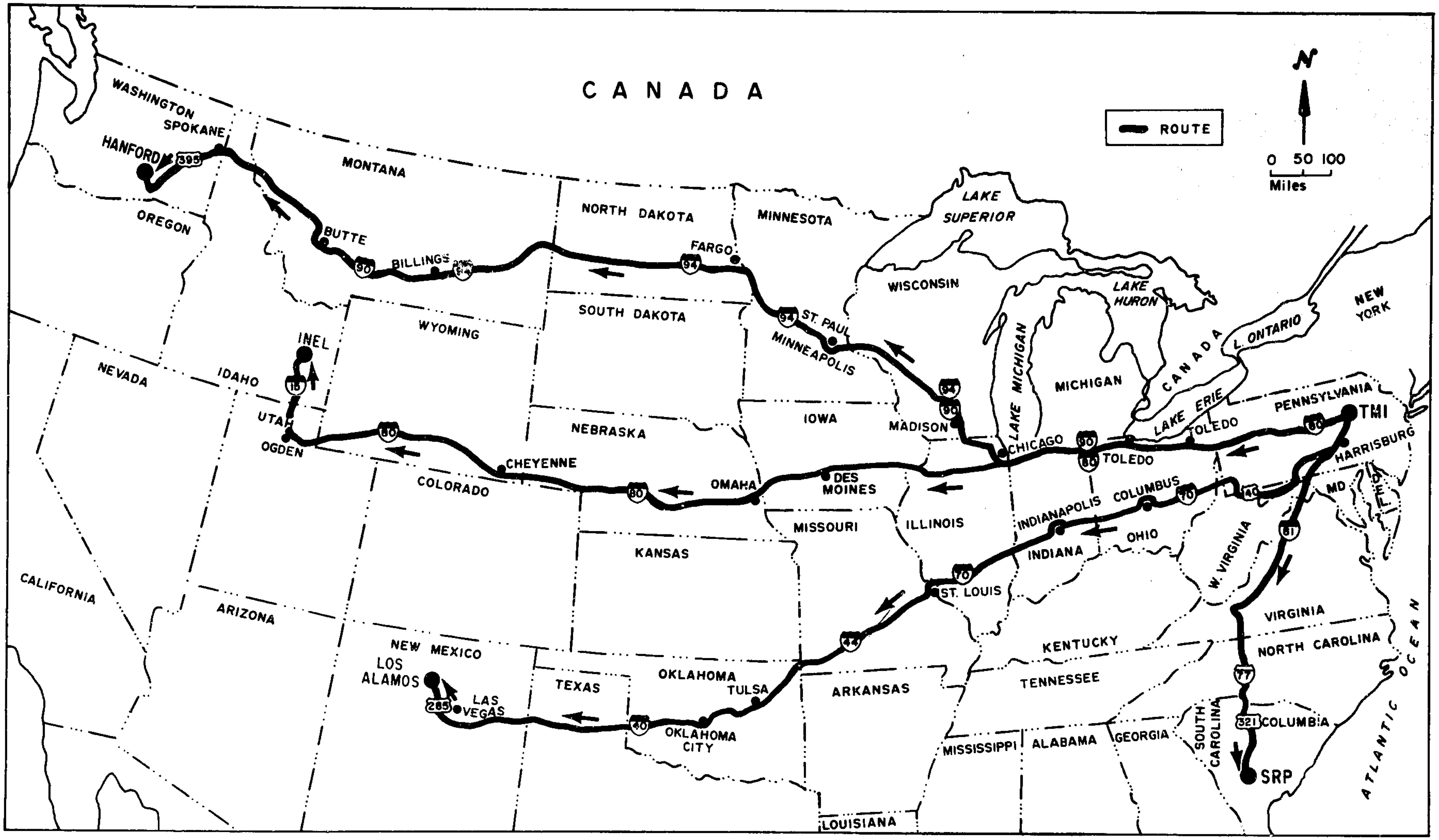
CANADA
ROUTE
N
0 50 100
Miles
WASHINGTON
SPOKANE
HANFORD
395
OREGON
MONTANA
BUTTE
90
BILLINGS
94
NORTH DAKOTA
FARGO
MINNESOTA
LAKE SUPERIOR
WISCONSIN
LAKE HURON
ST. PAUL
MINNEAPOLIS
MICHIGAN
LAKE MICHIGAN
CANADA
L. ONTARIO
NEW YORK
LAKE ERIE
PENNSYLVANIA
TMI
HARRISBURG
TOLEDO
MADISON
CHICAGO
80
IOWA
DES MOINES
SOUTH DAKOTA
WYOMING
INEL
15
IDAHO
NEVADA
UTAH
OGDEN
NEBRASKA
CHEYENNE
COLORADO
OMAHA
MISSOURI
ILLINOIS
INDIANAPOLIS
COLUMBUS
70
76
OHIO
INDIANA
MD
DEL
81
W. VIRGINIA
KANSAS
ST. LOUIS
44
CALIFORNIA
ARIZONA
NEW MEXICO
LOS ALAMOS
285
LAS VEGAS
TEXAS
OKLAHOMA
TULSA
OKLAHOMA CITY
40
ARKANSAS
LOUISIANA
KENTUCKY
TENNESSEE
MISSISSIPPI
ALABAMA
GEORGIA
SOUTH CAROLINA
VIRGINIA
NORTH CAROLINA
77
321
COLUMBIA
SRP
ATLANTIC OCEAN

Figure 9.3. Routes to DOE Treatment and Storage Facilities.

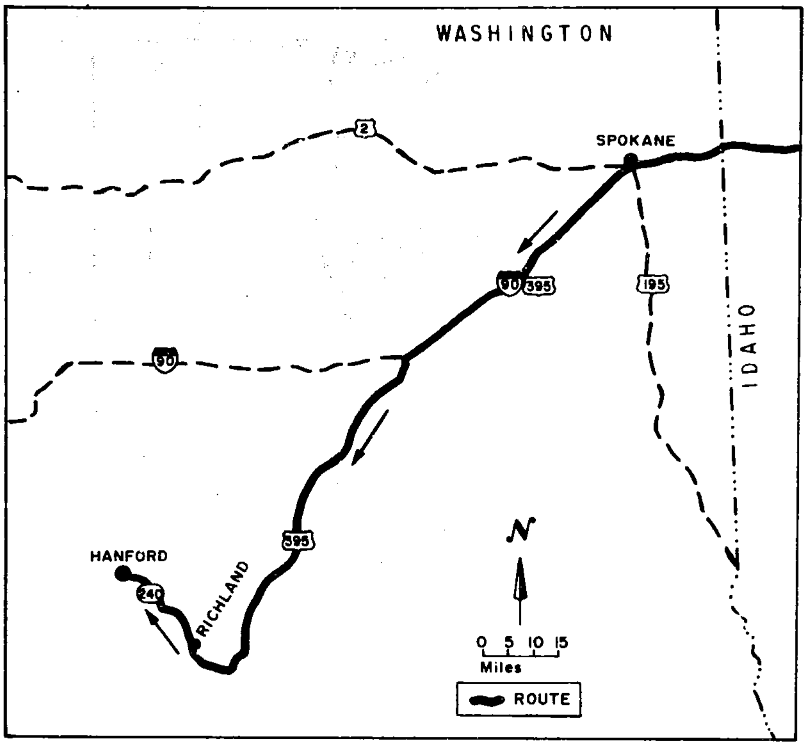
WASHINGTON
IDAHO
SPOKANE
2
90
395
195
90
395
240
HANFORD
RICHLAND
N
0 5 10 15
Miles
ROUTE

Figure 9.4. Routing in Washington to Hanford/Richland Facilities.

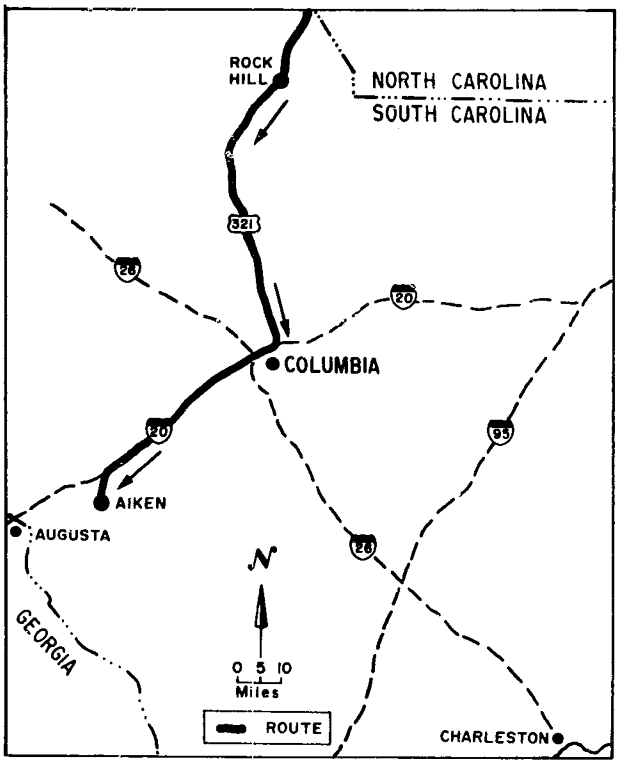
ROCK HILL
NORTH CAROLINA
SOUTH CAROLINA
321
26
20
COLUMBIA
20
95
AIKEN
AUGUSTA
GEORGIA
26
N
0 5 10
Miles
ROUTE
CHARLESTON

Figure 9.5. Routing in South Carolina to Savannah River Plant.

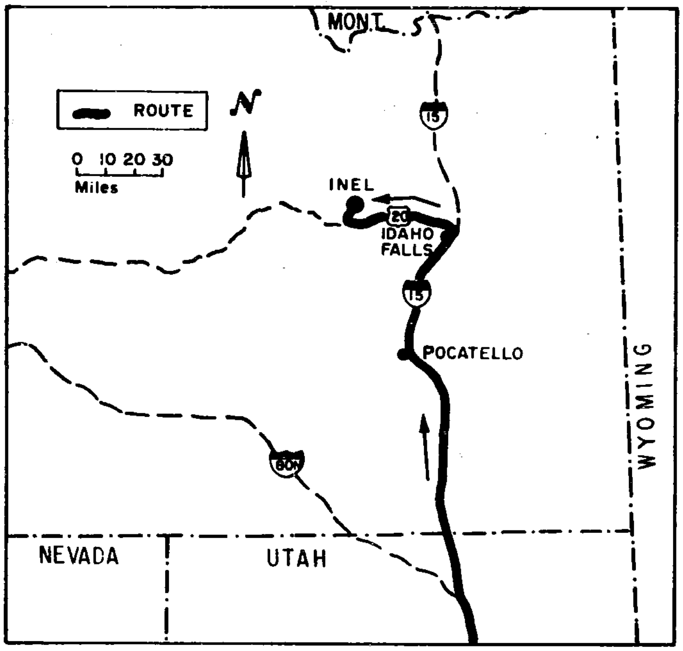
MONT.
ROUTE
N
0 10 20 30
Miles
15
INEL
20
IDAHO
FALLS
15
POCATELLO
80N
WYOMING
NEVADA
UTAH

Figure 9.6. Routing in Idaho to Idaho National Engineering Laboratory.

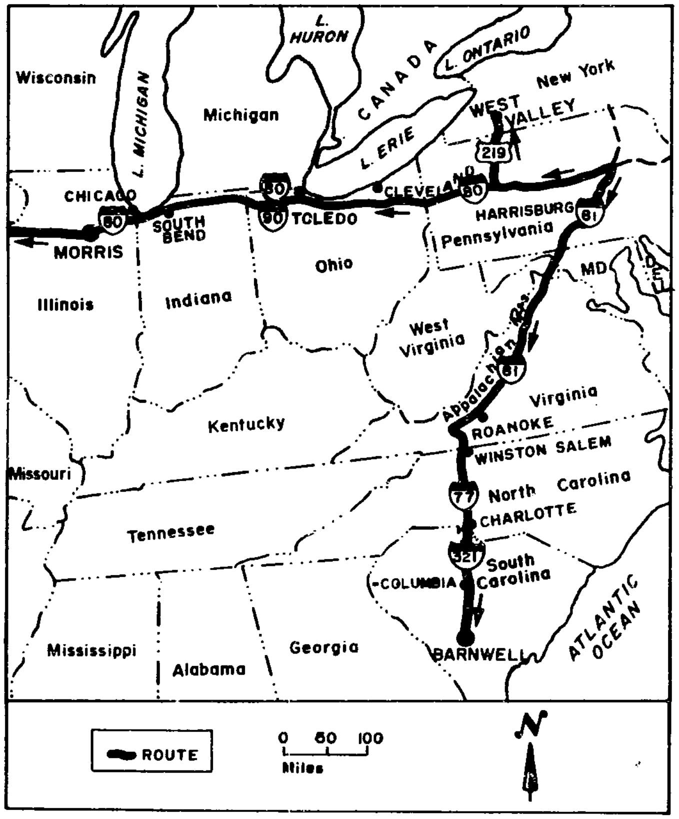
Wisconsin
L. MICHIGAN
Michigan
L. HURON
CANADA
L. ONTARIO
New York
WEST VALLEY
L. ERIE
219
CHICAGO
80
SOUTH BEND
80
90
TOLEDO
CLEVELAND
80
HARRISBURG
81
MORRIS
Pennsylvania
Ohio
MD
Illinois
Indiana
West Virginia
Appalachian Mts.
81
Virginia
Kentucky
ROANOKE
WINSTON SALEM
Missouri
77
North Carolina
CHARLOTTE
Tennessee
321
South Carolina
COLUMBIA
Mississippi
Alabama
Georgia
BARNWELL
ATLANTIC OCEAN
ROUTE
0 50 100
Miles
N

Figure 9.7. Routes to Potential Spent Fuel Storage Facilities.

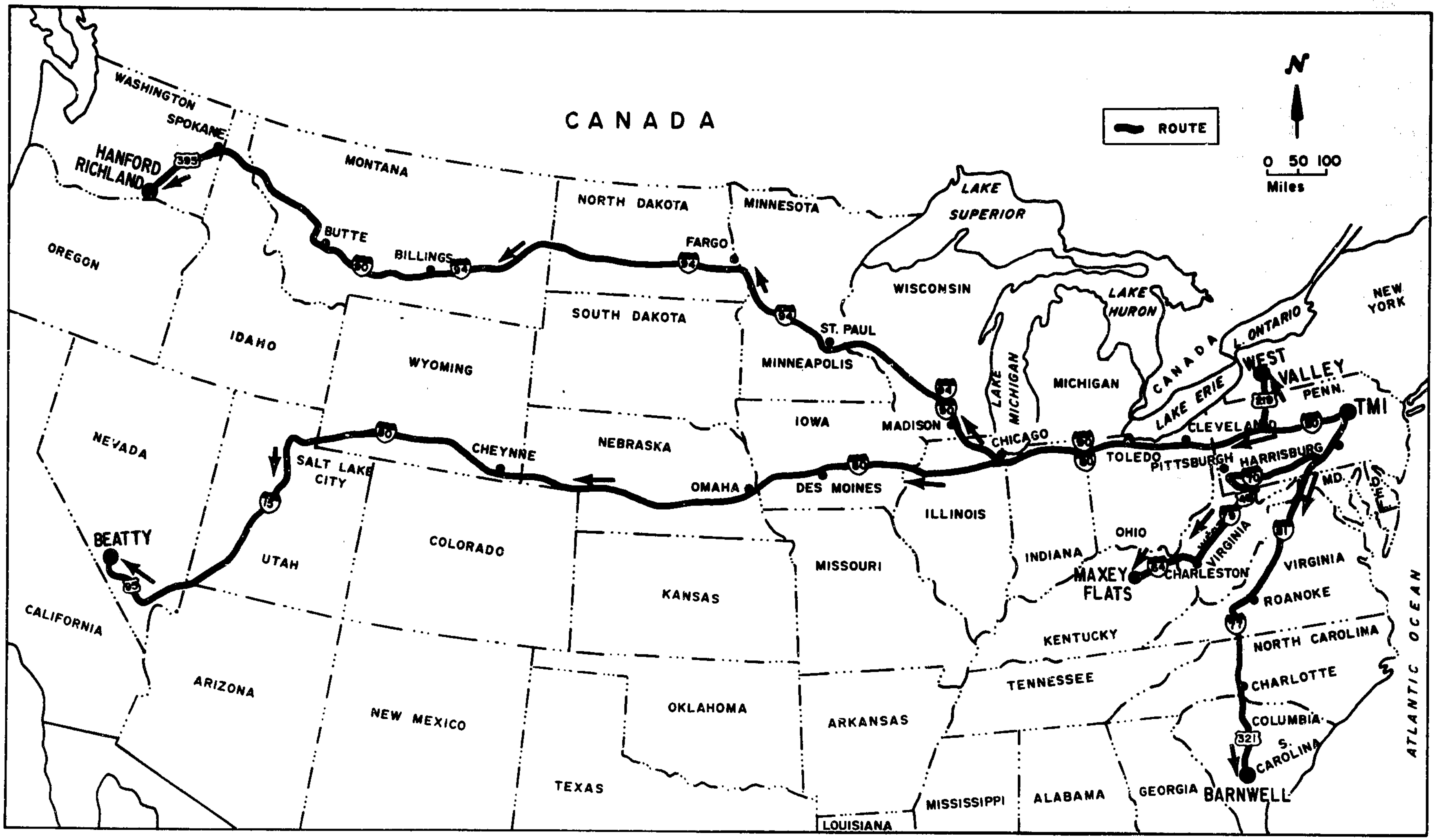
CANADA
ROUTE
N
0 50 100
Miles
WASHINGTON
SPOKANE
HANFORD
RICHLAND
395
OREGON
MONTANA
BUTTE
BILLINGS
90
94
IDAHO
WYOMING
NORTH DAKOTA
FARGO
SOUTH DAKOTA
MINNESOTA
MINNEAPOLIS
ST. PAUL
WISCONSIN
LAKE SUPERIOR
LAKE MICHIGAN
MICHIGAN
LAKE HURON
CANADA
L. ONTARIO
LAKE ERIE
NEW YORK
WEST VALLEY
219
PENN.
TMI
CLEVELAND
PITTSBURGH
HARRISBURG
TOLEDO
CHICAGO
MADISON
IOWA
DES MOINES
OMAHA
NEBRASKA
CHEYNNE
80
SALT LAKE CITY
15
95
NEVADA
BEATTY
CALIFORNIA
UTAH
COLORADO
KANSAS
MISSOURI
ILLINOIS
INDIANA
OHIO
MAXEY FLATS
64
CHARLESTON
W. VIRGINIA
MD.
DEL.
VIRGINIA
ROANOKE
81
77
NORTH CAROLINA
CHARLOTTE
COLUMBIA
321
S. CAROLINA
BARNWELL
KENTUCKY
TENNESSEE
ARIZONA
NEW MEXICO
OKLAHOMA
TEXAS
ARKANSAS
LOUISIANA
MISSISSIPPI
ALABAMA
GEORGIA
ATLANTIC OCEAN

Figure 9.8. Routes to LLW Disposal Facilities.

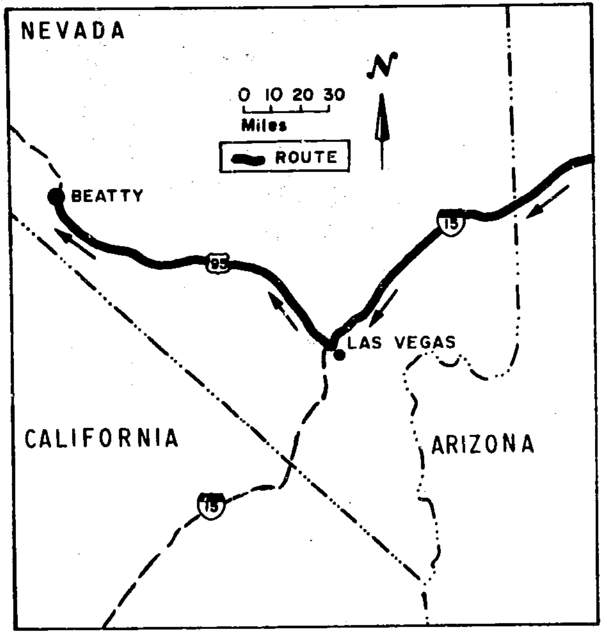
NEVADA
0 10 20 30
Miles
ROUTE
N
BEATTY
95
15
LAS VEGAS
CALIFORNIA
ARIZONA
15

Figure 9.9. Routing in Nevada to Beatty LLW Burial Site.

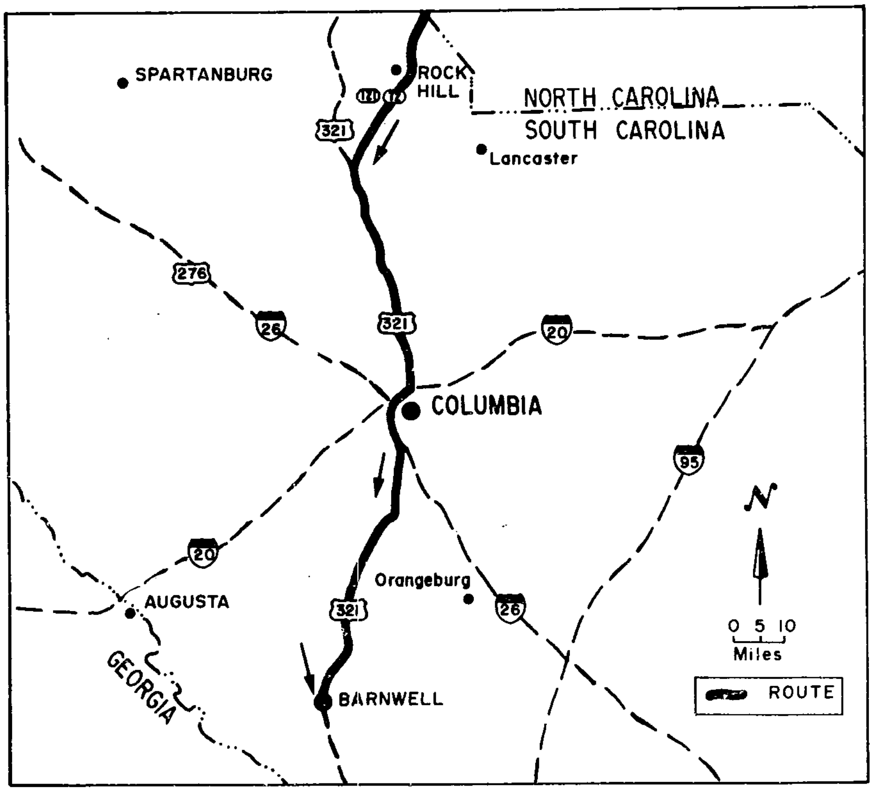
SPARTANBURG
ROCK HILL
NORTH CAROLINA
SOUTH CAROLINA
Lancaster
321
276
26
321
20
COLUMBIA
95
N
20
AUGUSTA
Orangeburg
321
26
GEORGIA
BARNWELL
0 5 10
Miles
ROUTE

Figure 9.10. Routing in South Carolina to Barnwell LLW Disposal Site.

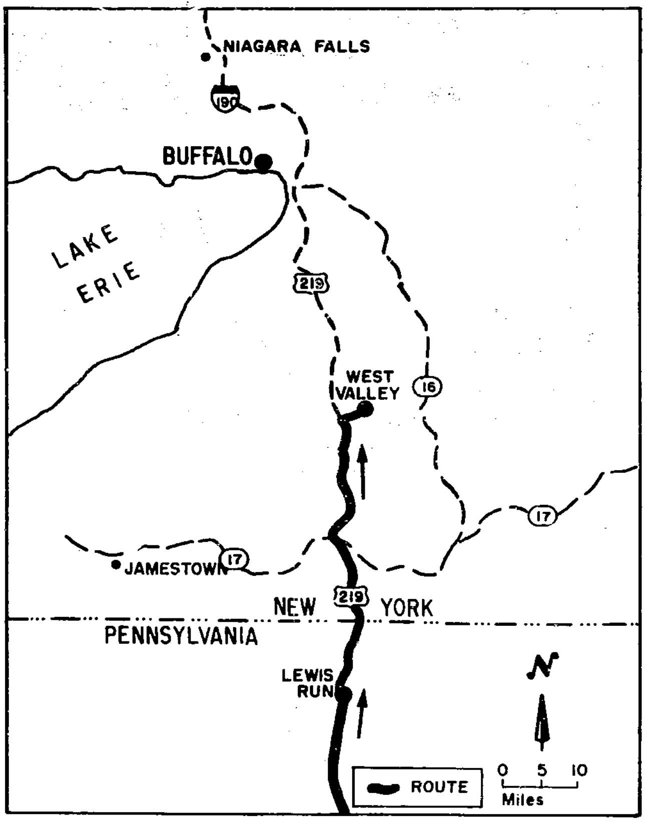
NIAGARA FALLS
190
BUFFALO
LAKE ERIE
219
WEST VALLEY
16
17
JAMESTOWN
17
NEW YORK
219
PENNSYLVANIA
LEWIS RUN
N
ROUTE
0 5 10
Miles

Figure 9.11. Routing in New York to West Valley LLW Burial Site.

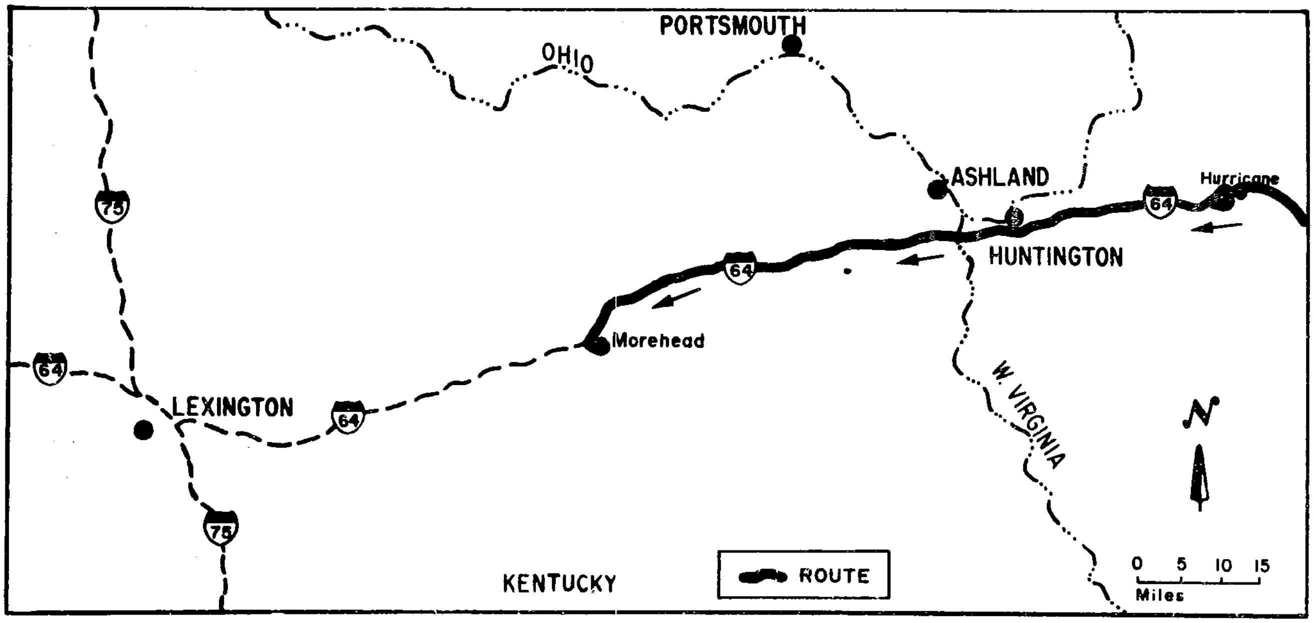
PORTSMOUTH
OHIO
ASHLAND
Hurricane
64
HUNTINGTON
64
75
Morehead
64
LEXINGTON
64
W. VIRGINIA
N
75
KENTUCKY
ROUTE
0 5 10 15
Miles

Figure 9.12. Routing in Kentucky to Maxey Flats LLW Burial Site.

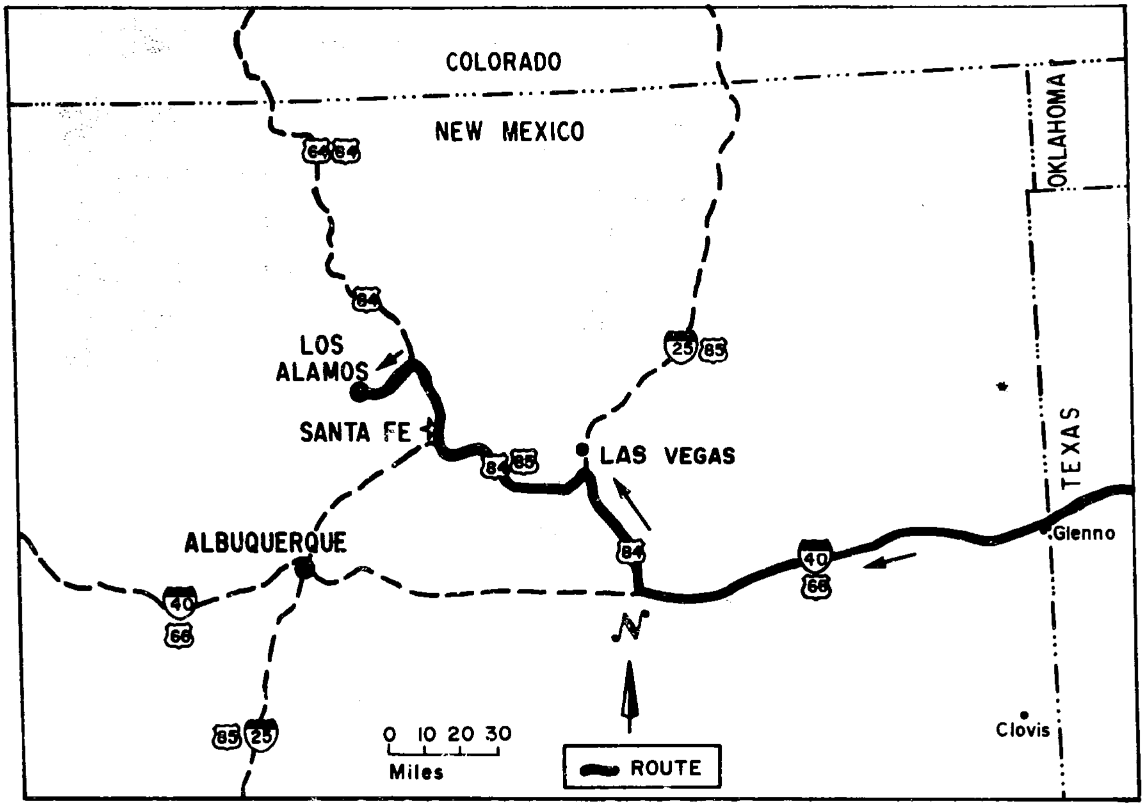

Figure 9.13. Routing in New Mexico to Los Alamos Site.

9.3.2 Disposal

LLW currently is being shipped from TMI-2 to the commercial burial site near Richland, Washington. Shipment of the remainder of the LLW to this site or one of the other commercial sites remains the preferred alternative, assuming current institutional constraints do not pose impediments. The alternatives of use of shutdown commercial sites or DOE sites, although operationally feasible, are less desirable because existing prohibitions would have to be lifted and new arrangements negotiated. This could be a lengthy and complex process.

In the case of HSAW, the use of intermediate depth burial, while a feasible disposal alternative currently being developed for this type of waste, is not a disposal technique in current practice at any burial site. The other option of disposal is a geologic repository. Since a geologic repository will not be in operation for 20 to 30 years, interim storage is a required step.

9.3.3 Transportation

Once the specific offsite storage, treatment, and disposal facilities have been identified, the appropriate truck transportation routes of those alternatives presented in Section 9.2.3 will be selected.

For the purpose of evaluating the potential bounds of the transportation impacts under normal and accident conditions, two feasible routes having extremes in distance traveled have been selected. These are:

- The current transportation route from TMI-2 to the commercial burial site at Richland, Washington, a distance of about 2750 miles.

The route to the shutdown waste management facilities at West Valley, New York, which is the closest feasible location for all the treatment, storage, and disposal alternatives. This distance is about 370 miles.

The rule proposed by DOT regarding the highway routing of radioactive materials (45 FR 7140, January 31, 1980) would make uniform laws by state and local governments on shipments of radioactive materials by highway. If adopted as proposed, existing state and local laws that prohibit the transportation of nuclear materials would be preempted. In effect, the proposed regulations establish the interstate highways as the preferred routes for movement of radioactive materials.

For the shipment of irradiated fuel, consideration must be given to this proposed DOT regulation, the NRC requirement for physical protection of shipments of irradiated fuel (10 CFR Part 73.37), and the NRC interim guidance for physical protection of such shipments (NUREG-0561, Rev. 1, June 1980). In consideration of these factors, the NRC has provided public disclosure of the selected shipment routes for irradiated fuel as well as information regarding safety and safeguards regulations (NUREG-0725, November 1980).

9.4 EFFLUENTS AND RELEASES TO THE ENVIRONMENT

Under normal storage, transportation, and disposal conditions, no routine effluents or releases from the waste packages or transport vehicles are expected. "Normal" transport is the situation when transport occurs without unusual delay, loss or damage to the package, or an accident involving the transporting vehicle.

The various types of packages that may be used to ship the TMI wastes are designed to prevent any releases during storage, handling, normal transportation, or disposal operations if their integrity is maintained. Thus, no impact on the environment from this source will occur.

The impact of the direct radiation from the packages under normal transport conditions is discussed in Section 9.5.1. Nonroutine releases under abnormal or accident conditions during transport are discussed in Section 9.5.2.

9.5 ENVIRONMENTAL IMPACTS

The principal environmental impacts from normal storage, transport, and disposal conditions are radiological, resulting from direct radiation exposure to package handlers, crew, and bystanders. Radiological impacts are discussed in the following sections for truck transportation from TMI to the storage or disposal site. In the case of accidents, direct radiological impacts on man are considered the most important potential component of postulated environmental impacts.

For this evaluation, the impacts are categorized under normal and accident conditions, and are further differentiated as occupational exposures to transport workers, environmental exposures to the population (bystanders), and nonradiological impacts.

9.5.1 Normal Conditions

For evaluation of the radiological impact in the case of normal storage, transport, and disposal, radionuclide content of the packaged waste and most characteristics of the package are not significant. The important characteristic is the radiation dose rate as a function of distance from the package surface. Federal regulations (DOT and NRC) impose constraints on the dose rate at specified distances from the package surface to protect package handlers, transport workers, and bystanders. Therefore, if required, storage areas and shipments of waste packages will be shielded or fenced to reduce the dose rate to acceptable levels. In addition, vehicles transporting the waste will be placarded to indicate to the public that they are transporting radioactive materials.

For this analysis, conservative dose estimates have been made by using the regulatory dose-rate limits for the TMI-2 shipments. In addition, where available, dose rates recorded during actual shipments ("experience" values) are used to obtain more realistic values. The material presented in NUREG-0170 and WASH-1238 has been the basis for much of the analysis presented in the succeeding sections.[6,7]

Estimated doses to handlers during storage, package movement, staging, and loading have been calculated by using experience values for exposure time for each step in the process in conjunction with expected exposure rates for the packages handled.

9.5.1.1 Occupational Doses

The occupational doses resulting from the storage, onsite movement, loading, and transport of the wastes are received by package handlers and vehicle crew members.

Onsite Storage and Handling

The onsite handling of unshielded packages from storage to truck loading involves the following steps: storage, pickup at storage area, transfer to loading area, placement in loading area, truck loading, and truck radiation survey.

In the case of shielded shipments, the packages will be placed in the shipping cask at the loading area and the shipping cask will be closed before the cask and truck are surveyed for radiation.

The analysis of the occupational doses for each different type of waste handling operation at TMI is presented in Appendix N. Using the relevant data from that appendix, the staff has determined the occupational radiation dose to a crew on a per-shipment basis and for all the waste shipments (best and worst case), including both shielded and unshielded packages. These data are summarized in Table 9.10.

Packaged waste will be removed from storage and loaded by two-member crews, and the loaded vehicle will be monitored by a radiation technician. A total of four two-member loading crews plus two technicians would be used for the shipments listed in Table 9.10.

The average dose to the loading crews was estimated based on an average of two shipments per day and time spans of 8 months for the minimum and 23 months for the maximum number of shipments. It was assumed by the staff that the radiation technicians would receive 10 percent of the dose and the loading crews 90 percent of the dose from each shipment. The resultant occupational doses are as follows:

- Radiation Technicians - For the minimum number of shipments each radiation technician would receive a total of 0.73 rem, or about 0.24 rem per quarter. For the maximum number of shipments, the total dose would be about 1.7 rem, or about 0.21 rem per quarter.

- Loading Crew - For the minimum number of shipments, each crew member would receive a total of 1.6 rem, or about 0.53 rem per quarter. For the maximum number of shipments, each crew member would receive a total of 4.6 rem, or about 0.58 rem per quarter.

Transportation

Occupational doses from truck transport of the wastes and spent fuel are received by the truck crew. The following parameters are assumed to apply to the shipments of the TMI waste:

- Shipments are made by exclusive-use vehicle.

- Dose rate is a maximum of 2 mrem/hr in any normally occupied position in the truck, and up to 10 mrem/hr at 6 ft from either an unshielded package or shielded cask.

- Two crew members per truck.

- Crew occupies cab of truck only during period of actual travel.

- The transport and shipment parameters presented in Table 9.11 apply to the TMI-2 waste shipments.[6]

- Two transport distances are used to bound the extremes of the dose analysis. The longest distance used is 2750 miles, the shortest is 370 miles.

The crew doses for the two routes considered are determined as follows:

Longest Route. Since the duration of crew exposure equals distance traveled divided by average speed, the total exposure duration is 60 hours. Assuming that a crew member spends 58 of

Table 9.10. Occupational Radiation Doses during Transfer of Waste Packages[e] from Storage and Truck Loading

Type of Shipment and Package	Cumulative Crew Dose[a] per Shipment (person-rem)	Best-Case Conditions		Worst-Case Conditions	
		Number of Shipments	Cumulative Total Crew Dose (person-rem)	Number of Shipments[b]	Cumulative Total Crew Dose (person-rem)
Unshielded shipments					
Drums[c]	0.15	27	4.1	128	19.2
LSA boxes[d]	0.046	104	4.8	177	8.1
Shielded drum shipments					
Sludge	0.017-0.06	6	0.4	7	0.4
Filter cartridges	0.003-0.006	1	Negligible	5	Negligible
Evaporator bottoms	0.017	0	-	119	2.0
Incinerator ash	0.018	34	0.6	-	-
Shielded ion-exchange material					
Liner shipments					
EPICOR II liners	0.016	69	1.1	69	1.1
Zeolite/resin system liners (reactor building sump)	0.024	8	0.3	33	0.8
Primary system cleanup resins (RCS)	0.016-0.024	8	0.2	68	1.6
Miscellaneous shielded shipments	0.022	40	0.9	208	4.6
Totals[e]		297	11.3	814	37.8

[a] Crew dose based on number of packages per shipment and exposure estimates in Appendix N.

[b] From Table 9.6, excluding damaged fuel assemblies.

[c] Unshielded trash drums include AFHB and RB decontamination solutions and compactable trash.

[d] Unshielded LSA boxes include noncompactable trash, contaminated equipment, irradiated hardware and mirror insulation.

[e] Exclusive of fuel packages. The occupational exposure during handling of fuel packages is included in the discussion of environmental impacts during core examination and defueling, Section 6.4.

Table 9.11. Transport Parameters for Calculation of Occupational Doses for Waste and Spent Fuel Shipments from TMI-2[a]

Parameters	High Population Areas	Medium Population Areas	Low Population Areas
Transport Parameters			
Average speed (mph)	15	25	55
Fraction of travel distance	0.05	0.05	0.9
Distance for TMI shipments (miles)			
- Longest route	138	138	2475
- Shortest route	19	19	333
Duration of exposure time (hrs)			
- Longest route	9.2	5.5	45.0
- Shortest route	1.3	0.8	6.1

[a] The number of shipments to all locations ranges from a minimum of 353 to a maximum of 997 (see Table 9.6).

those hours in the cab and 2 hours at a distance of 3 ft from the package (or cask), his/her maximum possible dose per trip is 182 mrem (2 mrem/hr* × 58 hr + 33 mrem/hr** × 2 hr). If the same crew member made 25 such trips per year, his/her annual dose would be 4.6 rem. The comparable annual cumulative crew dose would be 9.2 person-rem, and range from 130 person-rem to 360 person-rem for the range of shipments of all TMI wastes for the time interval required.

Shortest Route. In this case, the total exposure duration would be 8.2 hours. Maintaining the same proportions of the total hours in the cab and at 3 ft from the package results in a maximum individual dose per trip of 25.7 mrem and an annual individual dose of 0.8 rem for 30 trips. The crew cumulative dose would be 1.3 person-rem, and range from 18 person-rem to 51 person-rem for shipment of all TMI waste.

However, experience indicates[6] that dose rates in the cab of an exclusive-use truck are usually less than 0.2 mrem/hr, and about 25 mrem/hr at 3 ft from the package. On this basis the crew doses would be as follows:

Longest Route. The truck crew member would receive about 62 mrem per trip, and a maximum annual individual dose of 1.6 rem. For all shipments of TMI-2 waste the total cumulative occupational radiation dose would range from about 44 person-rem to about 124 person-rem.

Shortest Route. In this case, the truck crew member would receive about 9.1 mrem per trip, and a maximum annual individual dose of 0.2 rem. For all shipments of TMI-2 waste the total cumulative occupational radiation dose would range from about 6 person-rem to about 18 person-rem.

Offsite Storage and Disposal

The occupational doses to workers at the offsite storage sites are not treated here as additional impacts from the TMI-2 waste shipments; instead, such doses represent a component of the normal occupational doses associated with handling of all waste shipments at those facilities.

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*Dose rate in the cab.

**Dose rate at 3 ft from package.

9.5.1.2 Environmental (offsite) Exposures

Onsite Storage

The environmental exposures to the offsite population in the vicinity of TMI-2 resulting from onsite storage and handling operations are considered to be negligible. As previously noted, maximum radiation levels at the fence surrounding the facility from the interim storage and staging facility will be less than 0.6 mR/hr.

The nearest offsite location where members of the public can be expected to gather is the Visitors Center. It is conservatively postulated that there would be an average of ten people in the center for ten hours each day every day of the year. Under these conditions, the population dose would be about 0.0004 person-rem per year.

Transportation

Environmental exposures to the general population for truck transportation of the TMI waste involve exposure to people residing along the shipping route, exposure to persons in other vehicles on the route, and exposure to bystanders while the transport vehicle is stopped.

People Residing Along the Shipping Route

Longest Route. An estimated 700,000 persons who reside along a 2750-mile route from TMI to the disposal site might receive a cumulative population dose of about 0.05 person-rem for one shipment and a range of from 17 person-rem to 50 person-rem for all TMI waste and fuel shipments. These doses were calculated for persons in an area between 100 ft and 1/2 mile on either side of the shipping route, assuming 330 persons per square mile, 10 mrem/hr dose at 6 ft from the vehicle and each shipment traveling 200 miles per day.

Shortest Route. The estimated 125,000 persons residing along the 370-mile shortest route from TMI to the disposal site might receive a cumulative population dose of about 0.007 person-rem for one shipment and a range of from 2 person-rem to 7 person-rem for all TMI wastes and fuel shipments. The assumptions made are the same as for the longest route case.

The maximum dose received by a person living along any transport route, as determined in NUREG-0170,[6] would probably be received by an individual living adjacent to a highway where radioactive material was frequently shipped. The dose for one shipment received by a person living 100 ft from a roadway on which irradiated fuel shipments (worst case) would pass at an average speed of 30 mph would be 3.6×10^{-5} mrem. The total dose received by the same person for all TMI fuel shipments would range from 0.001 mrem to 0.004 mrem and, conservatively assuming the dose from a waste shipment to be the same as for the fuel, would range from 0.013 to 0.036 for all waste shipments.

Persons In Other Vehicles on the Route. The dose received by an individual driving 100 ft behind the truck (the location of probable maximum dose for persons sharing the transport route) for one hour would be 0.1 mrem. However, the staff believes that it is highly unlikely any individual driver of the general population would follow the truck for even such a long time, and instead would likely pass the truck or turn off after a short while.

Onlookers While Vehicle is Stopped. Members of the general public might be exposed to radiation from shipments of waste at points where the trucks stop along the route. A member of the general public who spends 3 minutes at an average distance of 3 ft from a loaded truck might receive a dose of up to 1.3 mrem. If ten people were so exposed during a shipment, the population dose for each shipment would be 0.013 person-rem, and for the total number of waste shipments from TMI, a population dose of from 5 person-rem to 13 person-rem might result.

Offsite Storage and Disposal

The environmental exposures to the offsite population adjacent to licensed storage and disposal sites are not treated as additional impacts from the TMI waste shipments. Instead, they are considered as a component of the normal doses associated with the handling of all waste shipments at that facility, regardless of origin. As such, these exposures are covered in the environmental impact statements for the specific sites.

9.5.1.3 Nonradiological Impacts

Nonradiological impacts on the environment from the normal storage, transport, and disposal of the TMI waste are primarily in the form of resource use. In the case of onsite storage and transport, the resource use takes the form of a commitment of packages and shipping containers to the TMI waste. The impact of such a commitment on the radioactive materials transportation industry is considered to be insignificant, except in those instances where specialized casks are required for the shipment of the irradiated fuel or other high-activity material requiring Type B packaging. In those instances, prior scheduling and coordination within the industry will be required to ensure availability of these packages on as-needed basis.

The major nonradiological impact will occur in the offsite storage and disposal of the waste packages. The packages that are disposed of by burial will use part of the limited capacity in the existing commercial low-level-waste disposal sites. If a burial efficiency of 50 percent is assumed for "random" placement of packages, based on current burial practices, burying the TMI packages will require from 243,000 ft^3 to 645,000 ft^3 of trench volume (see Table 9.12). For a typical trench 300 ft long by 30 ft wide by 20 ft deep, from two to four trenches will be needed for the TMI waste. Since the total remaining commercial waste disposal capacity is approximately 75 million cubic feet this will reduce the total commercial capacity available for disposal of wastes from other sources by 0.3 to 0.9 percent.

In the case of the intermediate level wastes that cannot be disposed of by normal shallow land burial, interim offsite storage may be used until final disposition is determined. The form of these wastes would be such that subsequent additional processing options for conversion into waste forms acceptable for the method used for final disposition would not be foreclosed. Assuming a storage efficiency of 75 percent based on current technology, storing the HSAW packages will require from 4000 ft^3 to 5100 ft^3 of storage capacity (see Table 9.13).

The irradiated fuel, if shipped to a reactor or AFR facility for storage, will utilize available capacity in a storage pool. For a range of 56 to 183 fuel elements, the cross-sectional area of storage pool space required will be from 66 ft^2 to 216 ft^2. The need for additional pool storage capacity will at some point exist for all AFR shipments for the nuclear power industry, with the TMI fuel elements representing only a very small increment of the total requirement.

9.5.2 Accident Effects

The other than "normal" occurrences associated with storage, transportation, and disposal can be categorized as either abnormal events or accidents. Abnormal events include occurrences that (1) compromise package integrity, such as dropping of packages by material handlers, packages being run over and crushed by a vehicle, and skewering of packages by a forklift, and (2) relate to packaging and handling procedures or package loss. Accidents are considered, in this context, to involve vehicles carrying the radioactive waste. Thus, abnormal occurrences are related to packaging and handling activities and accidents during transportation activities. The packaging and handling activities have been previously covered in Section 8; transportation accidents are covered in this section.

9.5.2.1 Transport Accident Probability

The probability of the occurrence of a transport accident that results in release of radioactive material can be described in terms of the expected number of accidents for the transport mode, together with the package response to those accidents and the dispersal that is expected.

Accident Rate

The accident rate used in this assessment for truck transport is 1.7×10^{-6} accident/mile (and for rail is 1.5×10^{-6} railcar accident/railcar mile).[6] Therefore, there is a probability of one accident occurring for every 314 truck shipments along the longest route (2750 miles) from TMI to a disposal site (worst case) and one for every 1590 truck shipments along the shortest route (370 miles) from TMI to the disposal site (best case). If all the waste is shipped on the long route, it is estimated that 1 to 3 accidents can occur for the range of waste shipments from TMI-2. On the other hand if all the waste is shipped on the short route, 0.2 to 0.6 accident can occur.

Table 9.12. Volumes of Packaged Solid Waste to Be Disposed of at a Commercial Low-Level Waste Disposal Site

		Best-Case Conditions			Worst-Case Conditions		
Type of Package	Package Volume (ft3)	Number of Packages	Shipped Volume (ft3)	Buried Volume[a] (ft3)	Number of Packages	Shipped Volume (ft3)	Buried Volume[a] (ft3)
55-Gallon Drums							
Low activity	7.5	3,200	24,000	48,000	15,400	115,500	231,000
Intermediate activity	7.5	502	3,765	7,530	1,707	12,800	25,600
LSA Boxes							
Low activity	80	1,042	83,360	167,720	2,128	170,240	340,480
Contaminated Equipment	70	86	6,020	12,040	293	20,510	41,020
and Hardware, Mirror Insulation	80	53	4,240	8,480	-	-	-
EPICOR II Resins							
1st stage[b]	50	49	2,450	4,900	49	2,450	4,900
2nd stage	50	14	700	1,400	14	700	1,400
3rd stage	175	6	1,050	2,100	6	1,050	2,100
RB Sump Cleanup							
Filters[c]	10	11	110	220	11	110	220
2nd stage	50	2	100	200	4	200	400
3rd stage	190	1	190	380	2	380	760
Primary System Cleanup[c]							
Filters[d]	10/7.5/150	16	990	1,980	57	1,340	2,680
2nd stage	50	4	200	400	44	2,200	4,400
3rd stage	190	3	570	1,140	12	2,280	4,560
Totals				256,520			659,520

[a] Assumes 50 percent efficiency.

[b] Will require special disposal procedures (e.g., deeper burial) if disposed of at a commercial disposal site.

[c] If any of these wastes contain fuel debris or greater than 10 nCi/gm transuranic materials, they would not be accepted at a commercial LLW facility.

[d] Primary system cleanup generates 3 filter types.

Table 9.13. Volumes of Waste Packages to Be Stored[a]

Type of Package	Number of Packages		Volume per Package (ft3)	Volume Shipped (ft3)		Total Volume of Storage Space Used[b] (ft3)	
	Best Case	Worst Case		Best Case	Worst Case	Best Case	Worst Case
SDS 1st stage	11	78	10	110	780	150	1,040
EPICOR II 1st stage[c]	46	46	50	2,300	2,300	3,070	3,070
Total						3,220	4,110

[a] Packages containing wastes from RCS cleanup that contain fuel debris or greater than 10 nCi/g transuranic materials also will require storage.

[b] 75 percent storage efficiency.

[c] To be stored if disposal at commercial burial sites with special precautions is not feasible.

Airborne Release Fraction for Accidental Release

To arrive at a realistic worst-case scenario for airborne releases under accident conditions, a severe truck accident sufficient to breach a Type B container with an attendant fire is postulated. As previously discussed, this accident is highly unlikely because the integrity of the Type B container has been demonstrated by engineering analysis and testing. Airborne releases would occur if the container ruptured and there was a fire or explosion. To determine airborne release fractions for the TMI waste forms, it is necessary to extrapolate data from other sources. In the absence of specific test data on TMI-2 wastes, release fractions for the TMI waste forms are assumed to be similar to fractions obtained from tests of simulated high-level waste immobilized in glass.8 It was found that after being subjected to container-rupture forces and fire, the fraction of the vitrified waste sufficiently small (less than 10 mm)* to become respirable was 5 v 10-5 for a container velocity of 55 mph on impact. It can be further assumed that only 20 percent of this fraction would actually be released from a Type B container,9 giving an overall respirable release fraction of 1 v 10-5. The staff believes this would be a bounding worst-case situation. The potential for airborne releases from representative Type B waste packages under combined impact (at a velocity of at least 50 mph) and fire is given in Table 9.14. The frequency of this specific type of accident has been estimated to be 6 v 10-13 accidents per truck mile, or 1.6 v 10-9 for shipments on the longest route, and 2.2 v 10-10 for shipments on the shortest route. This is the minimum accident that must occur to sufficiently breach a Type B package to cause a release of radioactivity.

Waterborne Release Fraction for Accidental Release

The accident scenario resulting in waterborne releases requires that a truck fall into a water body, the containers be punctured, and the waste forms be solubilized and carried out of the container and vehicle.

The solidified waste forms will be relatively insoluble in water, and it will take some time for water penetration and radionuclide mobilization to occur, permitting mitigating measures to be undertaken to minimize or prevent dispersion. However, if conservative release assumptions are made, a waterborne release fraction of about 1 v 10-6 is obtained for a breached package under prolonged leaking conditions. These assumptions include use of a solidified waste surface area

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*Particle size that becomes airborne and can be inhaled.

Table 9.14. Estimated Airborne Releases from Selected Type-B Waste Packages under Accident Conditions for 1×10^{-5} Fractional Release

Type of Package[a]	Package Content	Curie Content per Package	Release per Package (Ci)	Number Packages per Shipment	Release per Shipment (Ci)
Liners	Resins				
EPICOR II	1st stage	1,300	0.013	1	0.013
SDS zeolite	1st stage	120,000	1.2[b]	1	1.2
Drum	Filter	40	<0.001	14	0.006
	Evaporator bottoms	36	<0.001	14	0.005
	Accident sludge	435	<0.004	4	0.017
	Ash	2.0	<0.001	14	<0.001

[a] All shielded.

[b] Release of individual radionuclides per package for this worst case are: Cs-134, 0.168 Ci; Cs-137, 1.008 Ci; Sr-89, 0.0034 Ci; and Sr-90, 0.0144 Ci.

of 25 ft^2 before the accident, an area increase factor of 5 because of the accident, an average waste weight of about 1300 pounds per drum, a solubility of 1×10^{-6} cm^2-day, and five days of leaching. Applying this release fraction to an accident involving Type B packages results in the potential waterborne releases shown in Table 9.15. The frequency of such an accident would be less than one that would involve airborne releases, and would be dependent on the frequency and duration of proximity to water bodies during transport.

9.5.2.2 Transport Accident Radiological Risk

Once release of the radioactive material has occurred, radiological impacts would result from dispersion along a number of exposure pathways to man, including inhalation, ingestion, or direct radiation. Models have been developed to assess these impacts along the viable exposure pathways.[6] In addition, an accident may result in environmental contamination of land or structures, necessitating subsequent abandonment or cleanup operations.

The meteorological conditions that would exist if a transportation accident were to occur are difficult to predict since they would change with location and time. Relative to the severe truck accident described earlier in this section, the dose from inhalation was calculated assuming meteorological conditions equivalent to $\chi/Q = 5 \times 10^{-3}$ sec/m^3. The total body inhalation dose that would be received by an adult breathing at a rate of 8000 m^3/yr would be 100 mrem. This dose is for an individual assumed to be within several hundred feet for the duration of the accident. If the accident occurs near a vegetable garden, it may be necessary to take precautions to prevent large doses from vegetable consumption. The inhalation dose calculated represents a conservative value and may be lower depending upon meteorological conditions at the time of the accident.

9.5.2.3 Severe Accidents in Very High Population Density Urban Areas

Accidents occurring in urban areas of very high population density (> 10^4/km^2) may produce consequences more serious than discussed. This type of accident has a very small probability of occurrence which would be further diminished by the proposed new rules that would require that shipments of radioactive material be made on circumferential or bypass routes around urban centers.

Table 9.15. Estimated Waterborne Releases from Selected Type-B Waste Packages under Accident Conditions for 1 × 10^{-6} Fractional Release

Type of Package[a]	Package Content	Curie Content per Package	Release per Package (Ci)	Number Packages per Shipment	Release per Shipment (Ci)
Liners	Resins				
EPICOR II	1st stage	1,300	0.0013	1	0.0013
SDS zeolite	1st stage	120,000	0.12[b]	1	0.12
Drum	Filter	40	<0.0001	14	0.0006
	Evaporator bottoms	36	<0.0001	14	0.0005
	Accident sludge	435	<0.0004	4	0.0011
	Ash	2.0	<0.0001	14	<0.0001

[a] All shielded.

[b] Release of individual radionuclides per package for this worst case are: Cs-134, 0.0168 Ci; Cs-137, 0.100 Ci; Sr-89, 0.00034 Ci; and Sr-90, 0.0014 Ci.

9.5.3 Psychological-Socioeconomic Effects

Assuming two shipments per day, the transportation of solid waste would involve either 353 or 997 truck shipments over 14 or 34 months, depending on best- or worst-case conditions. The worst-case scenario for shipping solid waste would be modified further if a decision were made to transport the tritiated water offsite by truck; under this scenario, an additional average of 1.2 truck shipments per day would be made over a 16-month period.

Incident-free transportation of waste from the site is expected to produce little impact.[10] The current level of public sensitivity to and awareness of TMI-2 activities leads the staff to expect that the psychological and socioeconomic impact generated would be concentrated in the people living near the shipping route within the local impact area. The psychological impact is expected to be characterized as low-level anxiety, and would reflect apprehension of the possibility of accidents. The staff also believes that the transportation of waste through Middletown, where population density is higher and people live closer to the roadway than at other locations, in the vicinity of TMI, could result in the decreased marketability of residential property during the period of shipments. Shipments through Middletown also could result in adjustments in daily schedules and activities as households attempt to avoid the shipping route through Middletown.

The staff considered three accident scenarios involving offsite transportation: (1) incidents not involving environmental releases, (2) incidents resulting in releases to the air, and (3) incidents resulting in waterborne releases. Accidents involving airborne or waterborne releases are considered to be remote, although accidents not accompanied by releases to the environment are considered probable. If an accident does occur in the local area of impact, it would have the potential for producing a notable public response[10] because of current concerns focused on nuclear waste disposal. The nature and extent of any psychological or socioeconomic consequences are expected to be determined by the event, the factuality and duration of media coverage, the number of people impacted, and the danger posed to public health.

9.6 ECONOMIC COSTS

The incremental and total costs involved in the transportation of the solid waste and irradiated fuel from TMI, and the subsequent offsite storage, treatment (if applicable), and disposal charges are summarized in this section. In general the cost elements are (1) shipping cask use or rental, (2) transportation, and (3) commercial burial. The basic approach and methodologies used to quantify these cost elements and the incremental values are provided in Appendix K.

For the purpose of bounding disposal costs, the costs associated with both the maximum transit distance to Hanford (Richland), Washington, and the minimum distance to West Valley, New York, are determined for both the least and greatest number of shipments for each type of waste.

The range of estimated costs for transportation and storage of LLW and HSAW are between $2,610,000 and $6,680,000 for disposal at West Valley, New York, and $3,900,000 and $11,700,000 for disposal at Hanford, Washington.

Although it is recognized that there are cost factors associated with the processing and storage of the damaged fuel, sufficient data are not currently available to assess these considerations.

References--Section 9

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10. SUMMARY OF ENVIRONMENTAL IMPACTS OF THE PROPOSED ACTIVITIES

10.1 SUMMARY OF EFFLUENTS AND RELEASES TO THE ENVIRONMENT

In Sections 5 through 8 of this document, various alternative processes are described for each of the major stages in decontaminating TMI-2, for processing the radioactive wastes, and for preparing the wastes for shipment to offsite waste repositories. In Section 9 the procedures for transportation and terminal disposal of these wastes are described. For each of the cleanup activities, estimates were made of the potential releases of radionuclides for each alternative process. A detailed description of the effluents from each alternative process can be found in the appropriate sections. In some cases, a range of values representing the uncertainty of the estimates is indicated. The summary presented here is the summation of the amounts released from the process alternative that produces the greatest quantity of effluent of all processes considered for a particular activity. In cases where a range of values is given, the largest quantity is included in the summation.

Essentially all of the Kr-85 that was contained in the reactor building air (about 44,000 Ci) was vented to the atmosphere over the two-week period prior to July 11, 1980. There have been subsequent purges to release the Kr-85 that has been slowly desorbing from the water and walls. The monthly releases during September 1980 through December 1980 decreased in an approximately exponential manner (27, 15, 12 and 7.5 Ci). Future releases of radioactivity into the atmosphere will be a direct consequence of the normal decontamination processes, solidification and packaging of the wastes, and disposal of the fuel. During the normal decontamination operations, gases and very small particles of solid (dust) radionuclides will continually escape to the building atmosphere. The building ventilation systems contain HEPA filters that will trap most of the dust; however, some of the particles and all of the gases will pass through the filters and escape to the atmosphere. These effluents will accompany the operations throughout the entire period of the cleanup. The maximum estimates for the principal radionuclides which could be expected to be released during the course of the cleanup are summarized in Table 10.1.

During the course of the TMI-2 accident, a considerable quantity of water was contaminated with radionuclides as both dissolved and suspended solids. About 372,000 gallons of water flowed from the reactor system into the auxiliary and fuel handling buildings. This water has been processed by the EPICOR II system. About 700,000 gallons of contaminated water are in the bottom of the reactor building and will be processed during cleanup by one of three alternative means. The process chosen will also likely be used to process the nearly 96,000 gallons of water in the reactor coolant system. An additional source of contaminated water will result from the use of aqueous solutions or water sprays for decontamination of reactor building surfaces. This water also will need to be processed. The purpose of processing the water is to remove the suspended and dissolved radionuclides for subsequent immobilization and shipment to an offsite repository. However, the processed water will still contain some radionuclides, mainly tritium as HTO, which is not removed by the processing. Table 10.2 is a summation of the total radionuclide inventory remaining in all the processed water sources. The "best-case" and "worst-case" values represent the expected range of the residual radionuclide inventory in the processed water. This processed water must ultimately be utilized or disposed of by one of the means previously discussed in this document.

The discharge of processed water (tritium is the major remaining contaminant) to the Susquehanna River would occur only as a deliberate choice (and only after approval by the NRC) of that means of disposal. The processed water would be diluted with blowdown or river water and then discharged to the river at controlled rates such that the concentration of radionuclides in the river would be well below the threshold level for deleterious effects in aquatic species or humans. Table 10.2 is a summary of the total amount of the principal radionuclides that would be discharged under the above conditions.

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Table 10.1. Maximum Estimated Amounts of the Principal Radionuclides Released to the Atmosphere as a Consequence of Normal Cleanup Operations at TMI-2[a]

Document Section	Operation	Radionuclide Releases (Ci)[b]									
		H-3	Kr-85	Sr-89	Sr-90	Ru-106	Sb-125	Te-127m	Cs-134	Cs-137	Ce-144
5.1.4.1	Decontamination of the AFHB[c]	0.1	-	1.5×10^{-5}	9×10^{-5}	-	-	-	1.4×10^{-4}	7×10^{-4}	-
5.2.1.1, 5.2.4.1	Decontamination of the reactor building	750	44,000	-	5×10^{-6}	-	-	-	1.4×10^{-5}	8×10^{-5}	-
6.3.4.1, 6.4.4.1	Removal of RPV head and internals, core examination, and defueling	500	140	-	-	-	-	-	-	-	-
6.5.4.1,	Decontamination of primary system components	300	-	-	-	-	-	-	-	-	-
7.1.4.1	Liquid waste treatment[d]	0.3	-	7×10^{-4}	7×10^{-3}	-	-	-	8×10^{-3}	5×10^{-2}	-
7.2.4.1	Disposal of processed water[e]										
	Storage tank venting	0.1	-	-	-	-	-	-	-	-	-
	Evaporation from lined ponds	2900	-	-	-	-	-	-	-	-	-
	Forced evaporation in cooling tower[f]										
	Best case	2700	-	1.7×10^{-6}	3×10^{-5}	1.2×10^{-2}	2×10^{-2}	3×10^{-2}	$<7\times10^{-2}$	1.6×10^{-1}	7×10^{-3}
	Worst case	2700	-	1.7×10^{-1}	3	6	15	15	3×10^{-1}	1.4	1.5
8.1.4.1	Immobilization of process solid waste[g]	-	-	8×10^{-6}	8×10^{-5}	4×10^{-6}	3×10^{-7}	-	1.6×10^{-5}	1.1×10^{-4}	1.0×10^{-6}
8.2.4.1	Immobilization of chemical decontamination solutions[h]	-	-	-	7×10^{-8}	-	-	-	8×10^{-8}	6×10^{-7}	-
8.3.4.1	Packaging and handling solid materials[i]	-	-	1.3×10^{-3}	6×10^{-3}	-	-	-	3×10^{-2}	2×10^{-1}	-

Table 10.1. Continued

Document Section	Operation	Radionuclide Releases (Ci)[b] H-3	Kr-85	Sr-89	Sr-90	Ru-106	Sb-125	Te-127m	Cs-134	Cs-137	Ce-144
9.4	Storage, transportation, and disposal of fuel and solid waste[j]	-	-	-	-	-	-	-	-	-	-
	Total (for water disposal to river)[k]	1600	44,000	0.002	0.013	4×10^{-6}	3×10^{-7}	-	0.04	0.3	1×10^{-6}
	Total (for water disposal by natural evaporation)	2900[l]	44,000	0.002	0.013	4×10^{-6}	3×10^{-7}	-	0.04	0.3	1×10^{-6}
	Total (for water disposal by forced evaporation)[m]	2900[l]	44,000	0.17	3	6	15	15	0.3	1.7	1.5

[a] The releases for all radionuclides except H-3 and Kr-85 were calculated on the basis of an overall penetration factor of 0.001 for two HEPA filters in series (see Sec. 5.1.4.1).

[b] Values and totals are rounded to one, or in some cases two, significant digits.

[c] Effluents and releases prior to September 1, 1980, which were all below technical specification limits, are not included.

[d] See Table 7.11.

[e] The three entries under water disposal correspond to the three alternatives of: (1) storage and controlled release to the river; (2) natural evaporation; and (3) forced evaporation.

[f] See Table 7.27.

[g] See Table 8.17.

[h] See Table 8.37.

[i] See Tables 8.53 and 8.54.

[j] Under normal storage, transportation, and disposal conditions the waste and fuel remains sealed in containers so that no effluents or releases occur.

[k] This total corresponds to best-case water disposal conditions, i.e., disposal by controlled release to the river (for which airborne releases would be negligible) and worst-case conditions for all other operations.

[l] This is the total amount of H-3 in the accident water from the AFHB (370 Ci), reactor building (2500 Ci) and RCS (30 Ci). The individual releases for different operations add up to more than this total because mutually exclusive worst-case conditions were considered (e.g., tritiated water that evaporates from the spent fuel pool during defueling cannot also be released during liquid waste processing).

[m] The worst case (processing the water with the SDS alone) is assumed. Note, however, that release of effluent to the atmosphere must not result in offsite doses exceeding those proposed in Appendix R and discussed in Section 1.6.3.2.

Table 10.2. Amounts of Principal Radionuclides that Will Be Present in All of the Stored Water[a]

Radionuclides	Total Radioactivity in Processed Water (Ci)[b] Best Case (SDS/EPICOR II)[c]	Worst Case (SDS)[c]
H-3	2900	2900
Sr-89	6×10^{-6}	0.6
Sr-90	9×10^{-5}	9
Ru-106	0.04	21
Sb-125	0.07	54
Te-127m	0.1	51
Cs-134	<0.3	0.9
Cs-137	0.6	5
Ce-144	0.02	5

[a] The total volume of stored processed water would be slightly over 1.5 million gallons if no clean water were added and none was lost by evaporation. The origins of this water are: 743,000 gallons from the AFHB that has already been processed by EPICOR II, 700,000 gallons of contaminated water in the reactor building basement that has not yet been processed, and 96,000 gallons of water in the primary system of the reactor that also remains to be processed (see Tables 7.23 and 7.24). If the processed water were released to the river, the rate and the mixing with uncontaminated water would be adjusted so that the concentration of radionuclides in the river would be well below the threshold level for deleterious effects in aquatic species or humans.

[b] Values are rounded to one or two significant digits.

[c] See Section 7.1.3.3 for a discussion of these systems.

Except for the small releases to air and water (if approved), by far the greatest part of the radionuclides from the accident will be immobilized as solids and shipped offsite to authorized waste disposal facilities. It is very likely that the bulk of these shipments will be by truck as is presently the case. Although shielding is generally not necessary for truckloads of drums or low-specific-activity (LSA) boxes, high-specific-activity ion-exchange materials and spent nuclear fuel must be packaged in shielded casks for shipment. A summary of the estimated number of solid waste shipments, broken down by type of packaging, is presented in Table 10.3.

The potential accidents that could occur during cleanup operations are described in Sections 5 through 9 of this document. These accidents are postulated on very conservative assumptions, and the releases to the environment are similarly computed. The estimated amounts of the principal radionuclides that would be released to the atmosphere for each accident are listed in Table 10.4. For those accidents that would result in releases of radioactive materials into the river, the concentrations of radionuclides at the nearest intake for potable water would be a small fraction of the limits set forth in 10 CFR Part 20 for unrestricted releases. Details of each postulated accident and the effluent estimates can be found in the document section listed in the first column of Table 10.4. It is assumed by the staff that two or more accidents would not happen simultaneously because the cleanup activities with which the accidents are associated will take place at different times or at different places on the site, or both.

Table 10.3. Summary of Estimated Number of Waste Shipments[a]

Type of Waste	Number of Shipments Best Case	Worst Case
Low-level solids		
Drums	13	108
LSA boxes	104	177
Decontamination liquids		
Unshielded drums	14	20
Shielded drums	0	119
Shielded ion-exchange materials	85	170
Accident sludge, spent filters, ash	41	12
Contaminated equipment, materials, hardware	40	208
Damaged fuel assemblies	56	183
Total	353	997

[a] See Table 9.6 and accompanying discussion in Section 9.

10.2 SUMMARY OF OCCUPATIONAL DOSES AND HEALTH EFFECTS

Estimates for the occupational doses are presented and discussed individually for each of the major cleanup operations in appropriate subsections in Sections 5 through 9. Estimates of the duration of the work effort are included in the discussion. These estimates are summarized below.

10.2.1 Occupational Dose

The cumulative occupational radiation dose from all decontamination, defueling and waste disposal operations is estimated to be between 2000 and 8000 person-rem. Contributions from the different cleanup operations are summarized in Table 10.5. Defueling, primary system cleanup, and related operations contribute almost half of the total; decontamination of the reactor building contributes about one-third, and the remaining operations, which include decontamination of the auxiliary and fuel handling building, processing the contaminated water, and packaging and disposing of the waste, contribute the remainder.

10.2.2 Duration of Effort

The total time that will be needed to complete decontamination and defueling is estimated by the staff to be 5 to 9 years from the time of the accident on March 28, 1979. This estimate does not take into account delays due to financial problems, litigation, or other unpredictable events; the actual time could well be longer. The licensee's schedule calls for completion of cleanup activities in 1986, approximately 7 years from the accident (see Fig. 1.4).

Table 10.4. Amounts of the Principal Radionuclides Released to the Atmosphere as a Consequence of Postulated Accidents during Cleanup of TMI-2

Document Section	Accident	Radionuclide Releases (Ci)[a]							
		Kr-85	Sr-89	Sr-90	Ru-106	Sb-125	Cs-134	Cs-137	Cs-144
5.1.4.2	HEPA filter failure during decontamination of the AFHB[b]	-	1.5×10^{-5}	9×10^{-5}	-	-	1.4×10^{-4}	7×10^{-4}	-
5.2.4.2	HEPA filter failure during decontamination of the reactor building	-	-	5×10^{-6}	-	-	1.4×10^{-5}	8×10^{-5}	-
6.3.4.2 6.4.4.2	Release of trapped fission products during removal of the RPV head and internals, core examination, and defueling	70	-	-	-	-	-	-	-
6.5.4.2	Spill of decontamination liquids from reactor coolant system	-	8×10^{-5}	8×10^{-4}	-	-	1.6×10^{-4}	1.0×10^{-3}	-
7.1.4.2	HEPA filter failure during liquid waste treatment	-	7×10^{-4}	7×10^{-3}	-	-	8×10^{-3}	5×10^{-2}	-
8.1.4.2	Package handling accidents during immobilization of process solid wastes from:								
	EPICOR II	-	-	-	-	-	4×10^{-2}	2×10^{-1}	-
	Modified EPICOR II	-	4×10^{-2}	4×10^{-1}	8×10^{-3}	8×10^{-5}	8×10^{-2}	5×10^{-1}	3×10^{-3}
	SDS system	-	1×10^{-2}	1.2×10^{-1}	4×10^{-3}	4×10^{-4}	1.6	10	9×10^{-4}
	Other wastes[c]	-	5×10^{-4}	9×10^{-3}	1×10^{-4}	1×10^{-4}	4×10^{-3}	2×10^{-2}	7×10^{-5}

Table 10.4. Continued

Document Section	Accident	Radionuclide Releases (Ci)[a]							
		Kr-85	Sr-89	Sr-90	Ru-106	Sb-125	Cs-134	Cs-137	Cs-144
8.2.4.2	Package handling accidents during immobilization of chemical decontamination solutions[d]	-	-	1×10^{-8}	-	-	1.3×10^{-7}	9×10^{-7}	-
8.3.4.2	Storage area fire during packaging and handling of solid materials	-	9×10^{-4}	3×10^{-3}	-	-	6×10^{-3}	3×10^{-2}	-
8.3.4.2	Package handling accident while packaging, handling and storing solid waste[e]	-	6×10^{-6}	2×10^{-5}	-	-	4×10^{-5}	2×10^{-4}	-
9.5.2	Transportations accidents[f]	-	3×10^{-3}	1.4×10^{-2}	-	-	1.7×10^{-1}	1.0	-

[a] Values are rounded to one or two significant digits.

[b] Includes releases from surface decontamination and desludging (see Table 5.2).

[c] Other wastes include accident sludge, immobilized evaporator bottoms and bituminized decontamination liquids (see Table 8.20).

[d] Entries are the releases for an accident with a single package (see Table 8.38). A maximum of about 5000 packages will be handled (see Table 8.36).

[e] This is the release from a single package containing mirror insulation. The release from an accident with one of the other package types, which include drums of compactible trash or incinerator ash, or LSA boxes containing noncompactible trash, would be less (see Table 8.57). The number of packages handled would be in the range of 3000 to 15,000 (see Table 8.50).

[f] Releases are those expected for a transportation accident involving a type-B package containing first-stage SDS zeolite liners which would have a radioactivity content of 120,000 Ci per package (see Table 9.14). The number of packages of this kind that will be shipped should not exceed 90 (see Table 9.14).

Table 10.5. Summary of Cumulative Dose and Health Effects to Workers from Cleanup of TMI-2

Document Section	Operation	Cumulative Occupational Dose (person-rem)	Health Effects[a] Additional Cancer Deaths in Work Force	Health Effects[a] Additional Genetic Effects Among Offspring of Work Force
4.5.1	Maintenance of the Reactor in Safe Condition	8	0.001	0.002
5.1.5.1	Decontamination of the Auxiliary and Fuel Handling Buildings	375 - 550	0.05 - 0.07	0.10 - 0.14
5.2.5.1	Decontamination of the Reactor Building	660 - 3000	0.09 - 0.4	0.2 - 0.8
6.2.5.1	Reactor Coolant System Inspection	52 - 580	0.007 - 0.08	0.014 - 0.15
6.3.5.1	Removal of RPV Head and Internals	150 - 450	0.02 - 0.06	0.04 - 0.12
6.4.5.1	Core Examination and Defueling	580 - 1350	0.08 - 0.2	0.15 - 0.4
6.5.5.1	Decontamination of Primary System Components	108 - 1740	0.014 - 0.2	0.03 - 0.5
7.1.5.1	Liquid Waste Treatment	43 - 121	0.006 - 0.016	0.01 - 0.03
8.1.5.1	Handling and Packaging of Process Solid Wastes	17	0.002	0.004
8.2.5.1	Handling and Packaging of Chemical Decontamination Solution Wastes	3 - 10	0.0004 - 0.001	0.0008 - 0.003
8.3.5.1	Handling and Packaging of Solid Wastes	39 - 99	0.005 - 0.013	0.01 - 0.03
9.5.1.1	Transfer from Storage and Truck Loading	11 - 38	0.001 - 0.005	0.003 - 0.009
9.5.1.1	Transportation[b]	6 - 360	0.001 - 0.05	0.002 - 0.09
Totals		2000 - 8000[a]	0.3 - 1	0.5 - 2

[a] Values have been rounded to one or two significant digits; totals have been rounded to one significant digit.

[b] Different routes and different estimates for the expected exposure during transit lead to a large range in the transportation estimates; see Sec. 9.5.1.1.

The basis for the staff estimate is (starting from the date of issuance of this PEIS, approximately 2 years from the time of the accident): 18 to 48 months to decontaminate the reactor building to the point that defueling activities can begin; 3 to 10 months to remove the reactor pressure vessel head, upper internals and core support structure; 8 to 13 months for core examination and defueling; 3 to 12 months for primary system decontamination; and a final 6 months to complete interim waste storage or waste disposal activities. The large spread in the reactor building decontamination estimates reflects the bounding alternatives of: (1) proceeding simultaneously with sump water cleanup and building decontamination (and using lower bounds for the task duration estimates) or (2) deferring building decontamination until the sump water has been removed (and using upper bounds for task duration estimates). The large spread in the primary system decontamination estimates reflects the uncertainty in the condition of the primary system and the extent to which fuel debtis has been distributed throughout the system.

10.2.3 Health Effects

The work force for the TMI-2 cleanup will be exposed predominantly to penetrating radiation distributed over the whole body, so that any consequences will not be restricted to a particular area or organ of the body. A great deal of data on the biological (health) effects of radiation has been accumulated on a worldwide basis over the past several decades. These data have been analyzed by international and national organizations responsible for radiation protection.[1,2] The up-to-date findings of these organizations are the basis for estimating radiation-related human health effects in this document. The occupational doses from routine operations during the course of the TMI-2 cleanup may result in somatic and genetic effects. The somatic effect (to the body of the worker) of greatest concern is the possibility of inducing a fatal cancer; the genetic effects include a variety of inheritable changes which may affect future generations.

The risk factors utilized in the estimates of health effects are:

- 131 fatal cancers in the exposed workers per one million person-rem.
- 260 genetic effects among the offspring of the work force per one million person-rem.

More detailed information on the health effect risk estimators used by the staff is contained in Appendix Z.

It should be stressed that these risks, or probabilities, are increments above or additions to those risks to which the entire population currently is exposed. Current public health statistics show that for the entire U.S. population there is a 1 in 5 probability that death will be due to some form of cancer. The normal occurrence of hereditary disease in the offspring of the present U.S. population is about 1 in 17. It is expected that the occupational dose to the work force cleaning up TMI-2 will increase the workers' risk of death from cancer, but this added risk is relatively small in comparison with the existing risk. In addition, the risk of genetic changes can be expected to increase for the offspring of the work force, but this increment is also very small compared to existing risks.

The health effects from occupational exposure to radiation were calculated for the work force on the basis of radiation doses ranging between 2000 and 8000 person-rem. For the minimum cumulative dose case (2000 person-rem), it is expected that 0.3 additional fatal cancer would be caused. For the maximum dose case (8000 person-rem), 1 additional cancer fatality would result. Although it is possible to compute a range of probabilities for cancer induction among average individual workers based on the above figures, the results of such a calculation may not bear a close relationship to actual risks since the work force size and cumulative dose associated with the various tasks can differ by large factors, rendering inapplicable the concept of an average individual worker.

The licensee applies administrative controls for doses to its employees in order to ensure compliance with the regulations given in 10 CFR Part 20. These controls result in keeping most doses to less than 1 rem per quarter. Most of the workers involved in the cleanup can be expected to be in this category. 10 CFR Part 20 regulations limit the highest quarterly dose that an individual worker may receive to 3 rem per quarter. Individuals are not allowed to receive exposures in excess of 1 rem per quarter unless there are special circumstances. For example, a complex task that would normally be done by a single worker might require several workers if the 1 rem per quarter administrative control were imposed. In such situations, the total exposure to the work force can often be reduced if one worker is allowed to exceed 1 rem per quarter (but not the 10 CFR Part 20 limits) in order to complete the task.

For an individual worker who gets 1 rem per quarter throughout an assumed nine-year cleanup period, the total dose would be 36 rem. For a person of age 30 (the expected average age of cleanup workers), the probability of dying of cancer would normally be 1 in 5. The added probability of a premature death from cancer as a result of receiving a radiation dose of 36 rem would be 1 in 210. Thus, for the decontamination workers, the overall probability of death from cancer would be 1 in 4.9. The equivalent decrease in life expectancy from a 36-rem dose would be about 23 days. The risk for a younger worker would be greater, and for an older worker it would be less.

For the minimum cumulative dose case, the expected number of genetic effects among the offspring of the work force would be 0.5. For the maximum cumulative dose case, the expected number would be 2. The normal (exclusive of occupational dose) incidence rate for a hypothetical work force of 1000 persons would be about 60.

10.3 OFFSITE DOSES AND HEALTH EFFECTS FROM NORMAL OPERATION

For estimating dose and health effects, the quantities of radioactive material that may be released from the plant as a result of cleanup alternatives are based on the description of the source terms and the radioactive waste treatment system alternatives given in this statement. Site-specific and environmental data provided during licensing in the environmental report and in subsequent Metropolitan Edison answers to NRC staff questions relating to this PEIS were used extensively in performing dose calculations. Using estimated quantities of radioactive materials anticipated to be released and exposure pathway information, the dose commitments to individuals and to the population were estimated. Population doses from atmospheric releases are based on the projected population distribution within 50 miles in the year 2010 (3.2 million people). Population doses from liquid releases for consumption of drinking water are conservatively based on an assumed consumer population of 2.2 million people. The calculational methods used for estimating offsite doses given in this statement were those in Regulatory Guide 1.109.[3] Since the dose models in Regulatory Guide 1.109 assume a normally operating reactor with a 30-year lifetime, certain minor modifications were necessary to account for the relatively shorter time period of the decontamination program. These modifications and site-specific parameters are discussed in Appendix W.

The internal dose commitments in this statement represent the total dose received over a period of 50 years following the intake of radioactivity for one year under the environmental conditions existing during the cleanup period. This calculational approach, which is consistent with the recommendations of the ICRP, is described in detail in NUREG-0172.[4]

10.3.1 Dose Commitments from Radioactive Releases to the Atmosphere

Radioactive materials released to the atmosphere during the cleanup operation will result in small radiation doses to individuals and populations. NRC staff estimates of the expected gaseous and particulate releases (summarized in Table 10.1) and site meteorological considerations (discussed in Appendix W of this statement and summarized in Tables W-2 through W-6) were used to estimate radiation doses to individuals and populations. The results of the calculations are summarized below.

10.3.1.1 Radiation Dose Commitments to Individuals

The basis for selection of maximum individual receptor location (1.05 miles east of the site) and pathways considered for the maximum individual are described in Appendix W. The estimated dose commitments to the maximum exposed individual from tritium and particulate releases at the selected offsite location resulting in highest doses are listed in Table 10.6. Pathways from which radiation dose at this location resulted include inhalation, ground shine, cow milk ingestion, and vegetable consumption.

10.3.1.2 Radiation Dose Commitments to Populations

The estimated radiation dose commitments to the population within a 50-mile radius of the Three Mile Island nuclear plant from tritium and particulate releases to the atmosphere are shown in Table 10.7. Annual natural background radiation doses to the population within 50 miles would be 370,000 person-rem. The population dose commitments from atmospheric releases resulting from the cleanup operation represent an extremely small increase in the normal population dose due to background radiation sources.

Table 10.6. Dose Estimates for the Maximum Exposed Individual for Normal Radioactive Releases to the Atmosphere during Decommissioning and Decontamination of the Facility

Document Section or Table Number	Description	Dose (mrem)[a]		
		Total-Body	Bone	Liver
Decontamination				
Section 4.5.2	Kr-85 Releases	1.0×10^{-3}	1.0×10^{-3}	1.0×10^{-3}
Table 5.4	AFHB Decontamination and Desludging	5.2×10^{-3}	2.3×10^{-2}	5.3×10^{-3}
	RB Decontamination			
Table 5.12	Lowest tritium estimate	1.2×10^{-3}	1.5×10^{-3}	1.5×10^{-3}
Table 5.13	Highest tritium estimate	3.4×10^{-1}	1.5×10^{-3}	3.4×10^{-1}
Table 6.6	Removal of RPV Head and Internals	2.2×10^{-1}	0	2.2×10^{-1}
Table 6.8	Core Examination and Defueling	5.2×10^{-5}	3.0×10^{-4}	1.1×10^{-4}
Table 6.16	Decon. of Primary System Components	1.3×10^{-1}	0	1.3×10^{-1}
Table 7.14	Processing Accident and Decon. Water	3.9×10^{-1}	1.7	3.3×10^{-1}
Table 7.31	Option - Natural Evaporation of Processed Water[c]	(1.3)[b]	(0)	(1.3)
Table 7.32	Option - Forced Evaporation of SDS/EPICOR II Processed Water[c]	(1.5)	(1.3)	(2.7)
Table 7.33	Option - Forced Evaporation of SDS Processed Water[c,f]	(120.)	(480.)	(11.)
Table 8.22	Processing Solid Wastes	4.3×10^{-3}	1.9×10^{-2}	2.9×10^{-3}
Table 8.40	Processing Chemical Decon. Solutions	3.7×10^{-6}	1.6×10^{-5}	3.8×10^{-6}
Table 8.62	Trash Compaction of AFHB Wastes	2.6×10^{-4}	1.1×10^{-3}	3.7×10^{-4}
Table 8.63	Trash Compaction of RB Wastes	3.3×10^{-4}	1.5×10^{-3}	1.1×10^{-3}
Table 8.64	Trash Incineration	1.2×10^{-3}	5.3×10^{-3}	1.7×10^{-3}
Decontamination Totals				
Options that result in highest dose[d,f]		120.	480.	12.
Options that result in lowest dose[e]		0.8	1.8	0.7

(continued)

Table 10.6. Continued

Document Section or Table Number	Description	Dose (mrem)[a]		
		Total-Body	Bone	Liver
Decommissioning				
U.14	DECON activities	9.0×10^{-6}	4.5×10^{-5}	4.8×10^{-5}
U.23	SAFSTOR activities	1.0×10^{-5}	4.3×10^{-5}	4.0×10^{-5}
U.35	ENTOMB activities	8.9×10^{-6}	4.5×10^{-5}	4.8×10^{-5}
Decommissioning Totals:		2.8×10^{-5}	1.3×10^{-4}	1.4×10^{-4}

[a] The special location for which dose estimates are presented here is the nearest cow/garden location. The doses that are listed here are for the age group (adults, teenagers, children, infants) for which it was calculated to be highest. For this location, children's doses were highest.

[b] Values in parentheses represent values for options.

[c] Four methods were evaluated for disposal of processed water: ship offsite, natural evaporation, forced evaporation, and release to river. Atmospheric releases result from the natural and forced evaporation options only. Values appear in this table for both.

[d] Based on forced evaporation option for disposal of processed water, SDS processing assumed.

[e] Based on disposal of processed water by shipping offsite or release to river, and lower tritium range.

[f] This alternative results in doses which exceed the numerical criteria of 10 CFR Part 50, Appendix I, with only one pass through the water treatment system.

Table 10.7. Dose Estimates for the Projected Population in Year 2010 Residing within 50-Mile Radius of TMI Resulting from Atmospheric Radioactive Releases during Decontamination and Decommissioning of the Facility[a]

Document Section	Description	50-Mile Total-Body Population Dose (person-rem)
Decontamination		
4.5.2	Maintain Reactor in Safe Condition	3×10^{-2}
5.1.5.2	AFHB Decontamination and Desludging	2×10^{-2}
5.2.5.2	Decontamination of Reactor Building	
	Lowest tritium range	2×10^{-2}
	Highest tritium range	6.
6.3.5.2	Removal of RPV Head and Internals	4.
6.4.5.2	Core Examination and Defueling	2×10^{-3}
6.5.5.2	Decontamination of Primary System	3.
7.1.5.2	Processing of Water	2.
7.2.5.2	Option - Natural Evaporation Proc. Water[c]	(30)[b]
7.2.5.2	Option - Forced Evap. - SDS/EPICOR II	(30)
7.2.5.2	Option - Forced Evap. - SDS[f]	(400)
8.1.5.2	Processing Solid Wastes	2×10^{-2}
8.2.5.2	Processing Chemical Decon. Solutions	2×10^{-5}
8.3.5.2	Trash Compaction of AFHB Wastes	1×10^{-3}
8.3.5.2	Trash Compaction of RB Wastes	2×10^{-3}
8.3.5.2	Trash Incineration	6×10^{-3}
Decontamination Totals	Options resulting in highest dose[d]	400.
	Options resulting in lowest dose[e]	10.
Decommissioning		
U.3.4.2	DECON activities	6×10^{-5}
U.4.6.2	SAFSTOR activities	7×10^{-5}
U.5.4.2	ENTOMB activities	6×10^{-4}
Decommissioning Totals		1×10^{-3}

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[a] All values rounded to one significant figure. Pathways considered were external shine, ground shine, inhalation, meat consumption, vegetable consumption and milk consumption.

[b] Values in parentheses represent values for optional programs.

[c] Four methods were evaluated for disposal of processed water: ship offsite, natural evaporation, forced evaporation, and release to river. Atmospheric releases result from the natural and forced evaporation options only. Values appear in this table for both.

[d] Based on forced evaporation option for disposing of processed water, modified SDS processing assumed, and higher tritium range of Section 5.2.5.2.

[e] Based on disposal of processed water by shipping offsite or release to river, and lower tritium range of Section 5.2.5.2.

[f] This alternative results in doses which exceed the numerical criteria of 10 CFR Part 50, Appendix I, with only one pass through the water treatment system.

10.3.2 Dose Commitments from Radioactive Liquid Releases to the Hydrosphere

Liquid radioactive effluents released to the hydrosphere as a result of the Three Mile Island cleanup operation have been estimated to result in small radiation doses to both individuals and the population. NRC staff estimates of the expected liquid releases and the site hydrological considerations (discussed in Appendix W of this statement and summarized in Table W.7) were used to calculate anticipated radiation dose commitments to individuals and populations. The results of these calculations are discussed below.

10.3.2.1 Radiation Dose Commitments to Individuals

The basis for selection of individual receptor locations where fish were caught, drinking water was taken, and shoreline exposure was received is described in Appendix W. The estimated cumulative dose commitments to the maximum exposed individual from liquid releases at selected offsite locations are listed in Table 10.8.

Table 10.8. Dose Estimates for the Maximum Exposed Individual
for Normal Radioactive Releases to the River
Resulting from Disposal of Processed Water

Document Table No.	Description	Dose (mrem)[a]		
		Total-Body	Bone	Liver
Table 7.29	Option - Release Processed Water to River - SDS/EPICOR II Processing[b]	1.1	1.2	1.6
Table 7.30	Option - Release Processed Water to River SDS Processing[e]	9.8	23.	11.
Table 7.32	Option - Release of Blowdown from Forced Evaporation of Processed Water - SDS/EPICOR II Processing	7.7×10^{-1}	8.8×10^{-1}	1.1
Table 7.33	Option - Release of Blowdown from Forced Evaporation of Processed Water - SDS Processing[e]	7.0	16.	8.
Totals:	Option Resulting in Highest Doses[c,e]	9.8	23.	11.
	Options resulting in Lowest Doses[d]	1.5×10^{-3}	2.1×10^{-3}	2.3×10^{-3}

[a] The doses that are listed here are for the age group (adults, teenagers, children, infants) calculated to receive highest dose. The total-body doses are for adults, the bone doses are for children, and the liver doses are for teenagers.

[b] Four methods were evaluated for disposal of processed water: ship offsite, forced evaporation, natural evaporation, and release to river. Liquid releases result from forced evaporation method (blowdown) and from direct releases to the river. Values in this table appear for both.

[c] Based on direct release to river option, SDS processing assumed.

[d] Based on discharge of treated routine operational liquids (Sec. 4.5.2) and disposal of processed water by shipping offsite or by natural evaporation.

[e] This option results in doses that exceed the numerical criteria of 10 CFR Part 50, Appendix I, with only one pass through the water treatment system.

0.3.2.2 Radiation Dose Commitments to Populations

he estimated annual radiation dose commitments to an assumed population of 2.2 million drinking-ater consumers downstream of the Three Mile Island nuclear plant are shown in Table 10.9. ackground radiation doses to the same population would be 255,000 person-rem. The dose commitents from liquid releases from the Three Mile Island cleanup operation represent an extremely mall increase in the normal population dose due to background radiation sources.

0.3.3 Transportation of Radioactive Material

he transportation of TMI waste from the reactor to burial grounds is within the scope of the NRC eport entitled, "Environmental Survey of Transportation of Radioactive Materials to and from uclear Power Plants."[5] The estimated population dose commitments associated with transportation f fuels and wastes are listed in Table 10.10.

Table 10.9. Dose Estimates to the Population Residing Downstream of TMI Resulting from Normal Radioactive Releases of Processed Water to the River

Document Section	Description	Downstream Population Dose (person-rem)[a]
7.3.5.2	Option - Release Processed Water to River - SDS/EPICOR II Processing[b]	30
7.3.5.2	Option - Release Processed Water to River - SDS Processing[e]	900
7.3.5.2	Option - Release of Blowdown from Forced Evaporation of Processed Water - SDS/EPICOR II Processing	7
7.3.5.2	Option - Release of Blowdown from Forced Evaporation of Processed Water - SDS Processing[e]	600
Totals:	Options resulting in highest doses[c,e] Options resulting in lowest doses[d]	900 0.02

[a] All values rounded to one significant figure. Pathways considered were drinking water consumption and sport fish consumption.

[b] Four methods were evaluated for disposal of processed water: ship offsite, natural evaporation, forced evaporation, and release to river. Liquid releases result from the forced evaporation method (blowdown) and from direct releases to the river. Values in this table appear for both.

[c] Based on the option of direct release to river, SDS processing assumed.

[d] Based on discharge of treated routine operational liquids (Section 4.5.2) and disposal of processed water by shipping offsite or by natural evaporation.

[e] This option results in doses that exceed the numerical criteria of 10 CFR Part 50, Appendix I, with only one pass through the water treatment system.

Table 10.10. Dose Estimates to the Population Residing Along Transport Routes Enroute to Waste Burial Grounds - Decontamination and Decommissioning

Document Section or Table No.	Description	Population Dose (person-rem)[a]
Decontamination:		
Section 9.5.1.2	Dose to the public residing along transportation routes:	
	Longest route	20-50[b]
	Shortest route	2-7
Section 9.5.1.2	Dose to onlookers while vehicle is stopped[c]	5-10
Decommissioning:		
Section 2.2	Dose to public residing along transportation routes (longest route)	100
Totals:	Range	100-200

[a] All estimates rounded off to one significant figure.

[b] Range reflects different assumptions on length of route: the longest route estimate is based on 700,000 persons residing along the 2750 mile route and calculated for the persons in an area between 100 ft and ½ mile on either side of the route, assuming 330 persons per square mile, 10 mrem/hr dose at 6 ft from the vehicle and each shipment traveling 200 miles per day. The shortest route is based on the same assumptions as above except that 125,000 persons are assumed to reside along a 370-mile route.

[c] Assumes an individual spends 3 minutes at an average of 3 ft from the loaded truck and that 10 persons are exposed during each shipment.

10.3.4 Radiological Impact on Man

Based on the NRC staff's evaluation of the potential performance of cleanup alternatives, it is concluded that the cleanup can be accomplished within the dose design objectives of 10 CFR Part 50, Appendix I.* In Table 10.11, the calculated maximum individual doses for the entire cleanup program for each option are compared to the dose design objectives for atmospheric and liquid releases. The estimated doses in Table 10.11 represent the total dose commitment over the entire decontamination program, which will last several years, whereas the Appendix I values in the table are annual dose values. Upon comparing the estimated values in Table 10.11 to the design objective values, all options, except for forced evaporation of the SDS processed water and direct release of water that has received only one pass through the SDS system to the river, are estimated to meet the annual Appendix I design criteria and 10 CFR Part 20.

In Appendix I dose design objectives are defined as the as low as reasonably achievable (ALARA) requirements for normally operating reactors and were not intended to apply to the post-accident situation at TMI-2. Nevertheless, these objectives do serve as a basis for comparing the magnitude of potential cleanup releases to the potential radiological environmental impact of a normally operating reactor. Furthermore, as described in Appendix R, the staff proposed that the dose design objectives be applied as technical specification limits for the purpose of the decontamination operation. Even if the offsite doses are as high as the Appendix I design objectives, the maximum individual doses will still be relatively small when compared with either natural background doses (~116 mrem/yr) or of the dose limits specified in 10 CFR Part 20 (500 mrem/yr). As a result, the staff has concluded that there will be no significant radiological impact on man from the decontamination operation.

Table 10.11. Dose Estimates for the Maximum Exposed Individual for Entire Decontamination Program for Each Processed Accident Water Disposition Alternative, and Comparison with 10 CFR 50 Requirements

Option[b]	Dose (mrem)[a] (Total-body/max. organ) Estimated Value	Appendix I, 10 CFR 50 Value
Atmospheric Releases:		
1. Ship processed water offsite	0.8/1.8	-/15
2. Natural Evaporation to Atmosphere	2.1/2.0	-/15
3. Forced Evaporation to Atmosphere - SDS/EPICOR II Processing	2.3/3.4	-/15
4. Forced Evaporation to Atmosphere SDS Processing	120./480.	-/15
5. Release Water to River - SDS/ EPICOR II Processing	0.8/1.8	-/15
6. Release Water to River SDS Processing	0.8/1.8	-/15
Liquid Releases:		
1. Ship processed water offsite	.0015/.0023	3/10
2. Natural Evaporation to Atmosphere[c]	.0015/.0023	3/10
3. Forced Evaporation to Atmosphere SDS/EPICOR II Processing	0.8/1.1	3/10
4. Forced Evaporation to Atmosphere SDS Processing	7.0/16.	3/10
5. Release Water to River - SDS EPICOR II Processing	1.1/1.6	3/10
6. Release Water to River SDS Processing	9.8/23.	3/10

[a] The dose estimates represent the contribution of all decontamination programs for the entire cleanup.

[b] Dose estimates are listed separately for atmospheric releases and liquid releases, rather than adding them together, because different individuals are generally considered to be involved, and because the 10 CFR 50 dose objectives are different for each.

[c] A small fraction of water vapor may precipitate as rain over the Chesapeake Bay watershed.

Radiological doses to the general public from normal operation of the decontamination program may result in:

a. Late somatic effects in the form of fatal and non-fatal cancer in various body organs, following age- and organ-specific latency periods within the exposed population, and

b. Fatal and non-fatal genetic disorders in future generations of the exposed population.

Estimates of these health effects, which could occur randomly in an exposed population, are normally based on estimates of cumulative population dose expressed as person-rems (average dose X number of people receiving dose). Population health effect estimates presented in this statement reflect the total effect incurred by the population from all cleanup activities at TMI. In order to quantify individual risks, calculations are also made here for the maximum exposed individual. Absolute risk estimators of 135** deaths from latent cancer per 10^6 total-body person-rems in the exposed population and 258 cases of genetic disorders per 10^6 total-body person-rems in the future generations of the exposed population were derived from the 1972 BEIR report and the Reactor Safety Study (WASH-1400, October 1975). This derivation assumes a linear, nonthreshold dose-effect relationship at all sublethal dose levels.† Total body cancer risk estimators are used because they are larger than those for bone or liver.††

Using the above risk estimators for cancer deaths and genetic disorders, health risks as a result of releases from TMI were calculated for the population residing around TMI and for the maximum exposed individual. Health risks also were calculated for the population residing along the transport route to waste disposal grounds. The results of these calculations are described below.

Table 10.12 lists the expected number of cancer deaths or genetic abnormalities, designated as rates, for the 50-mile population of 2.2 million people around TMI as a result of decontamination activities. As these values are much less than 1 they suggest that it is very unlikely that a future cancer death occurs in the exposed population over the remaining lifetime of the population, or that a genetic abnormality occurs in the next 5 generations of the exposed population that could be associated with the clean-up operation. A better appreciation of the meaning of the numeric values in Table 10.12 can be gleaned by comparison of the rates of Table 10.12 to expected cancer death rates and genetic abnormality rates from causes other than TMI releases in the same population. For example, in 1976, about 20 percent of all deaths in the United States were due to cancer.[6] If those statistics are applied to the 2.2 million people living within 50-miles of TMI, the number of people in this population expected to die of cancer is 440,000. This number can be directly compared to the numeric rates in Table 10.12. This comparison indicates that the incremental chance of fatal cancer to the population or to an average individual in the population due to the decontamination activities is in the range of 1 chance in 400 million to 1 chance in 4 million, depending on the decontamination option that is chosen.

The same type of estimates for the population residing along the transport route (700,000 people) are listed in Table 10.13. A similar comparison of the numeric rates in Table 10.13 to the number of individuals in the population along the route who are expected to die of cancer (20 percent of 700,000 = 140,000) indicates that the incremental chance of fatal cancer to the population along the route or to an average individual along the route is 1 chance in 20 million due to shipments of decontamination wastes.

BEIR-I describes the normal incidence of diseases in which there is some evidence as being associated with a genetic component abnormality as 6 in 100, or 6 percent (p. 57, Table 4, of Ref. 2). In a similar fashion as was done for cancer death rates, the genetic abnormality rate due to

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*Appendix R describes proposed modifications to the plant's operating technical specification which will impose the 10 CFR Part 50 Appendix I design objectives as operational limits.

**This value, 135, is slightly greater than the risk estimator (131) used for occupational dose effects because of age distribution differences in the two populations. The occupational population excludes individuals less than 18 years of age.

†Details of the derivation of radiation-induced health impacts on man are provided in Appendix Z.

††The risk estimates are 7 deaths and less than 22 deaths per 10^6 person-rem for bone and liver, respectively.

Table 10.12. Estimates of Cancer Death Rate and Genetic Abnormality Rate for Exposure of the Population around TMI due to Releases from Decontamination for the Range of Processing and Disposal Alternatives[a]

Decontamination Option	Estimated Dose (person-rem total body)	Rate[b] (deaths or abnormalities per 2.2 million people)	
		Cancer Fatality over Remaining Lifetime	Genetic Abnormality over Next 5 Generations
Ship processed water offsite	10	1×10^{-3}	3×10^{-3}
Forced evaporation of - SDS processed water	1000	1×10^{-1}	3×10^{-1}

[a] All estimates rounded off to one significant figure.

[b] Population death or abnormality rate estimates are expected to be slightly smaller than the values presented here because the year 2010 population estimate of 3.2 million people was used for dose estimates from atmospheric pathways.

Table 10.13. Estimate of Cancer Death Rate and Genetic Abnormality Rate for Exposure of the Population Residing along Transport Routes

Activity	Range of Dose (person-rem)[b]	Rate[a] (deaths or abnormality per 700,000 people)	
		Cancer Fatality over Remaining Lifetime	Genetric Abnormality over Next 5 Generations
Decontamination	20-50	7×10^{-3}	1×10^{-2}
Decommissioning	100	1×10^{-2}	3×10^{-2}

[a] Rate estimates based on upper limit of range.

[b] All values in table have been rounded off to one significant figure.

the decontamination and decommissioning activities at TMI can be compared to the incidence of diseases related to genetic abnormalities from causes other than TMI releases. For the 2.2 million people in the 50-mile radius around TMI the expected incidence of non-TMI related genetic abnormalities is 132,000 ($0.06 \times 2.2 \times 10^6 = 132{,}000$). Comparing this to the rates of Table 10.12 indicates that the incremental chance of a genetic abnormalities to descendants of this population or to an average exposed individual in it over the next five generations ranges from 1 chance in 40 million to 1 chance in 400 thousand, depending upon the decontamination options which are used.

In a similar fashion, for the 700,000 people residing along the transportation route it is estimated that the expected incidence of diseases related to genetic abnormalities from causes other than transport of TMI waste is 42,000 ($0.06 \times 700{,}000 = 42{,}000$). Comparing this to the rates of Table 10.13 indicates that the incremental chance of a genetic abnormality in descendants of the population or in descendents of an average exposed individual over the next five generations is 1 chance in 4 million due to radiation exposure along the transport route.

The total-body dose to the maximum exposed individual near TMI for the option involving shipping processed water offsite was estimated to be 0.8 mrem. The maximum individual dose is about 600 times larger than the average individual 50-mile population dose for the option involving shipping of water offsite. The incremental chance of cancer death to the maximum exposed individual was estimated to be 1 chance in 2 million, and the incremental chance of a genetic abnormality to the descendants of the maximum exposed individual in the next five generations was 1 chance in 300,000. If some other option is used for processing and disposal of the water the resulting maximum individual dose will be limited to the requirements of 10 CFR Part 50, Appendix I, on an annual basis, which would result in an increased chance of cancer death of 1 chance in 100,000, and an increased chance of a genetic abnormality in the next five generations of 1 chance in 20,000.

On the basis of the small likelihood of increased incidence of cancer deaths or genetic abnormalities from decontamination and decommissioning activites, the staff has concluded that the risk to the public health and safety will be insignificant.

10.4 DOSES FROM POSTULATED ACCIDENTS

10.4.1 Dose Commitments from Accidental Radioactive Releases to the Atmosphere

The quantities of radioactive material that may be released from postulated accidents are derived based on the systems described in this document and assumed failure scenarios. The licensee's site and environmental data provided in the environmental report and in subsequent answers to NRC staff questions relating to this PEIS are used extensively in the dose calculations. These quantities of radioactive materials released and the exposure pathway information are used to estimate the dose commitments to individuals. The calculational methods that were used throughout this statement for estimating offsite doses were those of Regulatory Guide 1.109.[3] Since Regulatory Guide 1.109 is for a normal operating reactor, certain minor modifications were made to take into account the short time period of accidental release. These modifications are discussed in Appendix W.

The dose commitments from accidents postulated in this statement represent the total dose received over a period of 50 years following the intake of radioactivity for one year under the environmental conditions existing during the cleanup period. For the younger age groups, changes in organ mass with age after the initial intake of radioactivity are accounted for in a stepwise manner.

10.4.1.1 HEPA Filter Failure during Decontamination Operations

During decontamination operations, dust and radioactive particles can become airborne and enter the building ventilation systems. To minimize releases of radioactive particles to the environment, the building air is drawn through the plant ventilation system. The ventilation exhaust system consists of a prefilter, HEPA filter, adsorber (charcoal), and final HEPA filter. The prefilter provides a measure of protection for the HEPA filters installed in the exhaust ventilation system. The HEPA filters are highly efficient for all particle sizes. A minimum efficiency of 99.97 percent for installed HEPA filters is based on the most penetrating particles (0.3 mm). The first HEPA filter in the system collects most of the radioactive material. If the first

HEPA filter failed, the second filter would collect most of the materials released from the first filter. The dose estimates summarized in Table 10.14 are based on the assumption that both filters fail simultaneously. For most operations, even if both filters failed simultaneously, doses would be within decontamination operation requirements (Appendix R). If only the first HEPA filter fails, the expected dose would be much smaller than those listed in the table and would be within regulatory criteria of 10 CFR Part 50, Appendix I, for normal operating reactors. Even if both filters failed, the resulting offsite doses for all accidents would be within requirements of 10 CFR Part 20.

Table 10.14. Dose Estimates for the Maximum Exposed Individual Caused by a HEPA Filter Failure during Specific Decontamination Operations

Table Number	Decontamination Operation	Dose (mrem)[a]		
		Total-Body	Bone	Liver
5.5	Surface Decontamination Operation in AFHB	4.2×10^{-3}	1.8×10^{-2}	1.4×10^{-2}
5.6	Desludging Operations in AFHB	2.0×10^{-1}	8.5×10^{-1}	1.9×10^{-1}
5.14	Decontamination of Reactor Building	1.3×10^{-2}	5.9×10^{-2}	2.2×10^{-2}
7.15	Processing Reactor Building Sump Water	3.3	16.	12.
7.16	Processing Reactor Building Decontamination Water	5.2×10^{-4}	2.3×10^{-3}	1.8×10^{-3}
7.17	Processing Chemical Decontamination Water	9.7×10^{-4}	4.2×10^{-3}	3.2×10^{-3}
7.18	Processing RCS Accident Water	1.5	6.0	2.8×10^{-1}
7.19	Processing RCS Flush and Drain Water	7.5	30.	1.3
7.20	Processing RCS Decontamination Water and RCS Chemical Decontamination Water	1.5	6.2	2.8×10^{-1}

[a] Doses were calculated for total-body, GI-tract, bone, liver, kidney, thyroid, lung and skin. The maximum three organ doses are listed in this table. Doses were calculated for four age groups; adults, teenagers, children, and infants. The dose estimates are for the nearest cow/garden location (1.05 miles east). The dose estimates presented in this table are for children.

For the following reasons, the probability of both HEPA filters failing simultaneously is very small. The most likely cause of HEPA filter failure would be overpressurization. During operation of the system, overpressurization can occur due to heavy buildup of particulate material on the filter. However, each HEPA filter in the ventilation exhaust system is provided with pressure indication. High pressure differential across a filter will set off local and remote alarms which would alert the operators to take action, such as shutting the system down.

Even if no action were taken to alleviate a HEPA filter overpressurization and the filter failed, the particulate matter that would come off the failed filter would collect on the second downstream HEPA with a removal efficiency of 99.97 percent. For there to be a significant release to the environment, both HEPA filters in the ventilation system would have to fail simultaneously.

Since a failure of a single filter in the series will not result in doses in excess of regulatory criteria for normal operating reactors (10 CFR Part 50, Appendix I) or in excess of those described in Appendix R of this statement, and since the chance of a multiple failure is highly unlikely, the staff concludes that the HEPA filter failure accident scenario does not pose a significant level of risk to the public health and safety.

10.4.1.2 Contaminated Material Fire

Most of the contaminated material onsite is water, sludge, ion-exchange media, and building and equipment surface materials that will not support combustion. However, there are large quantities of rags, blotter paper, plastic sheets, etc., used for decontamination activities. When these materials become contaminated, they are usually collected in plastic bags, compacted in steel drums, and placed in temporary storage prior to offsite shipment for disposal. There are also significant quantities of non-compactible contaminated materials such as lumber, tools, pump casings, etc., which are collected in wooden boxes and placed in temporary storage prior to offsite shipment for disposal. A fire in the storage area containing packaged drums and boxes is considered a credible accident. An analysis of fires in barrels of contaminated trash in a fuel reprocessing plant indicates that the fraction released from such fires is 8×10^{-7} of the total activity present. Using this model, the largest release analyzed (Section 8.3.4.2) is approximately 3×10^{-8} μCi with the radionuclide distribution listed in Table 8.53. The estimated dose resulting from this release to the maximum exposed individual is 6.5 mrem for total-body, 29 mrem for bone, and 8.5 mrem for liver. This accident involving a fire in a low level waste storage area does not result in offsite doses in excess of the requirements of 10 CFR Part 20. On this basis, the staff concludes that this type of accident does not pose a significant risk to the public health and safety.

10.4.1.3 Breach of a Waste-Containing Package

The staff has evaluated the consequences of breaching a package containing spent ion exchange media, spent filters, compactible trash, noncompactible trash, mirror insulation, or incinerator ash from a postulated drop of the package and its contents. The dropping of a waste package and the resulting breach and release of a portion of the contents of the package, is considered a credible accident. The estimated doses to the maximum exposed individual resulting from this type of accident for a variety of waste-containing packages are listed in Table 10.15. With the exception of the accident involving the postulated drop of a spent liner generated from the processing of primary system water through modified EPICOR II, none of the breach accidents listed in Table 10.15 results in offsite doses in excess of the requirements of 10 CFR Part 20, and only a few are above normal requirements described in Appendix R. In order to mitigate the consequences of a postulated drop of a modified EPICOR II zeolite liner used for processing primary system water, the licensee will be required to either administratively control the curie inventory on the zeolite liner or design and test, prior to actual use, a liner which will be capable of withstanding, without breaching, the worst-case accident (i.e., the highest drop onto an unyeilding surface) in the event modified EPICOR II is approved for processing primary system water. On this basis, the staff concludes that none of the breaching accidents listed in Table 10.15 poses a significant risk to the public health and safety.

10.4.1.4 Vehicle Accident Along Transport Route

To arrive at a realistic "worst-case" scenario for airborne releases resulting from a truck transport accident, the staff assumed that a Type B container would break preceding a fire of the package. Data from other sources suggest that a release fraction of 10^{-5} could be expected from such an accident. The total-body dose from such an accident under assumed meteorological conditions was estimated to be 100 mrem. The estimated dose indicates that this accident will not exceed the requirements of 10 CFR Part 20 and probably not exceed those of 10 CFR Part 50, Appendix I, as the assumed meteorological conditions are considered upper bound conditions. On the basis of these results, the staff concluded that this accident scenario will not result in significant impact to the environment or risk to the public.

10.4.1.5 Spill of Reactor Coolant System Liquid in Reactor Building

The staff has evaluated the consequences of spilling reactor coolant system (RCS) water in the reactor building while the primary system pumps are operating. For the purpose of estimating consequences, 10 percent of the liquid was assumed to spill in the reactor building (2000 Ci). About 0.1 percent of this would become airborne in the reactor building atmosphere and about 0.1 percent of the airborne fraction could be expected to pass through the HEPA filters. The resulting dose to the maximum exposed offsite individual was estimated to be 1.5 mrem to the total body, 6 mrem to the bone, and 0.26 mrem to the liver. These values are well below the requirements of 10 CFR Part 50, Appendix I, for normal operation and of 10 CFR Part 20. On this basis, the staff concludes that this type of accident does not pose a significant risk to the public health and safety.

10.4.2 Dose Commitments from Accidental Radioactive Releases to the Hydrosphere

Accidental radioactive effluent releases to the Susquehanna River that may occur during the cleanup operation are described in Section 7.2 in this statement. Site hydrological consideration for accidental releases are similar to those for normal decontamination releases which are presented in Appendix W. The differences are discussed in the sections of this document where the actual calculations are presented. The results of the postulated accident are summarized in Table 10.16. The type of accident that was considered in this document involves the failure of a processed water storage tank and subsequent leakage of its contents into the east channel of the Susquehanna River. This accident and its effects are discussed below.

Table 10.15. Dose Estimates for the Maximum Exposed Individual Caused by Breaching a Package Containing Radioactive Waste

Table Number	Type of Waste	Dose (mrem)		
		Total-Body	Bone	Liver
8.23	SDS-Reactor Building Sump Water Zeolite	8.1×10^{-4}	3.7×10^{-3}	2.8×10^{-3}
8.24	SDS-Filter Assembly	5.7×10^{-3}	2.2×10^{-2}	3.1×10^{-4}
8.25	SDS Cation Liner	1.6×10^{-3}	6.4×10^{-3}	1.5×10^{-6}
8.26	SDS Mixed-Bed Liner	1.2×10^{-1}	4.6×10^{-1}	1.5×10^{-4}
8.27	EPICOR II-AFHB; Prefilter Liner	16.	76.	88.
8.28	EPICOR II-AFHB; Cation Liner	1.6	6.3	6.5
8.29	EPICOR-II-AFHB; Mixed-Bed Liner	3.9×10^{-2}	1.7×10^{-1}	2.0×10^{-1}
8.30	Modified EPICOR II-Primary Water; Zeolite Liner	700	2900	790
8.31	Modified EPICOR II-Primary Water; Cation Liner	3.7	15.	4.6×10^{-3}
8.32	Modified EPICOR-II Primary Water, Mixed-Bed Liner	1.7×10^{-1}	6.7×10^{-1}	2.0×10^{-4}
8.41	AFHB & Reactor Building Chemical Decontamination	6.8×10^{-5}	3.0×10^{-4}	2.3×10^{-4}
8.66	Compactible Trash	3.0×10^{-2}	1.3×10^{-1}	4.5×10^{-2}
8.67	Noncompactible Trash	1.6×10^{-2}	6.6×10^{-2}	2.2×10^{-2}
8.68	Mirror Insulation Trash	4.6×10^{-2}	2.0×10^{-1}	6.5×10^{-2}
8.69	Compactible Ash	3.0×10^{-3}	1.3×10^{-2}	4.5×10^{-3}

Table 10.16. Dose Estimates for the Maximum Exposed Individual Caused by Breaching a Processed Water Storage Tank and Releasing Contents into the East Channel of the Susquehanna River

Table Number	Processing Option	Dose (mrem)[a] Total-Body	Bone	Liver
High River Flow[b]:				
7.34	SDS	0.95	2.2	1.1
Low River Flow:				
7.35	SDS/EPICOR II	56/kg fish consumed	-	-
7.35	SDS	470/kg fish consumed	-	-

[a] Total-body dose estimates are for adults, bone dose estimates are for children, and liver dose estimates are for teenagers.

[b] "High river flow" for the purposes of this table is defined as that river flow which causes overtopping of Red Hill Dam. "Low river flow" is defined as that river flow which does not cause Red Hill Dam to overtop.

10.4.2.1 Failure of Processed Water Storage Tank

During water processing operations, processed water will be temporarily stored in two holding tanks located outdoors, each of 500,000 gallon capacity. The water in the holding tanks will then be disposed of by one of the methods described in Section 7.2. If one of these tanks ruptured and its entire contents were released, storm drains would transport the water to the east channel of the river. The potential offsite dose to humans from this accident is highly dependent upon whether or not Red Hill Dam is overtopping. If Red Hill Dam is overtopping, the released water will be diluted with the flow of the Susquehanna River resulting in doses to humans that are fairly low and within annual limits for routine operation (see Appendix R). However, if Red Hill Dam is not overtopping, the released radioactivity could remain in the east channel for an extended period of time at fairly high concentrations. Dose calculations are presented in Table 10.16 for both river flow situations.

For the high river flow situation, the resulting offsite doses are estimated to be below the requirements of 10 CFR Part 20, and below the dose design objectives for normal operating reactors of 10 CFR Part 50, Appendix I. Thus, the staff concluded that if this postulated accident were to occur during high river flow, the resultant environmental impact would be insignificant.

For the low river flow situation, the resulting offsite doses due to consumption of drinking water or fish from the east channel would be large enough to warrant that action be taken to avoid such consumption. Consequently, the staff recommends that mitigative action be taken to avoid consumption of fish or drinking water from the east channel if the accident occurs during low river flow. Since there is no municipal use or known private use of water from the east channel for consumption purposes, doses are not expected to occur through the drinking water pathway. The main concern is to prohibit the catching and consumption of fish from the channel. As the bioaccumulation of radionuclides in fish occurs over periods of days to weeks, the staff concluded that there would be ample time to take preventive measures to ensure that fishing is stopped in the east channel area.

It is conceivable that a fish could reside in the channel for a long period, bioaccumulate radionuclides, and then move to some other area and be caught and consumed. Depending upon the water processing option, 1 to 10 kilograms (kg) of fish would need to be consumed before the 10 CFR Part 20 protective dose limits are exceeded. Assuming that the average weight of a fish harvested from the river near TMI is about 0.5 kg of whole body weight (thus yielding $\leq$ 0.2 kg of edible meat), an angler would have to harvest between 5 and 50 fish to obtain 1 to 10 kg of edible fish meat. Six years of studies have shown that the mean harvest from the York Haven Pond during the summer-fall months is less than one fish per angler (or per fishing trip by an angler). It seems unlikely, therefore, that any given angler would harvest enough fish (all of which had resided in the east river channel following a tank rupture) to permit consumption of enough meat to result in a dose that exceeds the protective limits. Additionally, studies of the post-accident (1979) river fishery showed that anglers released their catches in greater than normal proportions and ate fewer fish due to their concerns that the fish might have been radioactively contaminated by the accident. Similar angler behavior could be expected following a tank rupture, with adequate public notification, thus reducing the likelihood of any anglers receiving unacceptable doses from consuming river fishes.

Hence, with proper mitigative action, the public health and safety will be protected in the event of an accident during low flow conditions. Such mitigative actions could include fishing advisories or consumption bans; or physically blocking the movements of fish into and out of the shallow east river channel by placing a fine-mesh net across the channel near Sand Beach Island or the north access bridge.

10.4.2.2 Leakage of Reactor Building Sump Water

The largest amount of contaminated water presently on the site is the 700,000 gallons of water in the bottom of the reactor building. This water contains an estimated 500,000 Ci of radionuclides. It is postulated that if this water should leak through the thick steel-lined concrete base of the reactor building into the ground, it would ultimately reach the Susquehanna River. In Appendix V the movement of this water through the ground was analyzed. For purposes of this analysis, it was assumed that the volume of water above the water table could leak into the groundwater in one to two days. It was calculated that the water would not begin to reach the river for about one year, and while the water was in the ground, processes of adsorption, filtration, and ion exchange would remove a large fraction of the dissolved and particulate radionuclides except H-3. Dilution of the water reaching the river would further reduce the concentration of radionuclides so that the peak concentrations of Cs-137 in the river would be 5.1×10^{-10} μCi/mL, of Sr-90 would be 5.1×10^{-8} μCi/mL, and that of H-3 would be 5.2×10^{-7} μCi/mL. These values are orders of magnitude below the MPC limits of 10 CFR Part 20. Monitoring wells around the reactor building would provide indication of the increase in radionuclides in the groundwater long before they would reach the river; the remaining water could be transferred to storage tanks or other mitigating measures when a grout curtain could be instituted.

10.4.2.3 Accidents Associated with Dropping Heavy Components on the Vessel Seal Ring

At this stage of the operation, a drop accident that caused a failure of the seal plate would result in drainage of the transfer canal to the reactor building basement, but would have no significant radiological consequences since the canal water would contain very little radioactivity (0.01 μCi/mL, exclusive of tritium), and there would be no fuel in the transfer canal.

10.5 POTENTIAL RELEASES DUE TO EXTERNAL EVENTS

10.5.1 Potential Releases due to Flooding

An evaluation has been made to determine the potential impact of Susquehanna River floods on the nuclear waste storage facilities at Three Mile Island. Of primary concern are the interim storage facility (which is to be decommissioned shortly) and the concrete storage facility located in the Unit 2 desilting basin. These facilities were designed so as not to be affected by the design basis flood, but both would be inundated by the probable maximum flood (PMF). A description of the design basis flood and PMF is provided in Section 10.5.1.2. The design of the facilities has been reviewed to evaluate the potential for release of radioactivity to the environment during a PMF.

10.5.1.1 Temporary Radwaste Facility

The temporary radwaste storage facility was designed to provide shielded storage for 28 spent EPICOR I and II liners (16 cells 4.5 ft in diameter by 8 ft high, and 12 cells 7 ft in diameter by 8 ft high). The cells are galvanized, corrugated metal cylinders that have steel plates welded to one end to provide a bottom. The cells are placed on compacted earth fill in the Unit 2 desilting basin and backfilled with compacted earth. Each cell is provided with a 16-ton concrete shield cover (Appendix D). The elevation of the top of the cells is below the flood control dike elevation of 304 ft Maximum Sea Level (MSL) at the downstream side of the island. Any flood level exceeding the elevation of the dike would result in water levels greater than the top of the cells, but the cells are sealed against leakage by the welded base plate and the massive, sealed concrete top.

10.5.1.2 Interim Radwaste Storage Facility

The interim radwaste storage facility is a modular structure, each module consisting of a 57 × 91 × 19-ft-high, reinforced-concrete box with 3-ft-thick concrete base and 4-ft-thick walls (Appendix D). The top of the walls is at an elevation of 305.33 ft MSL. Each cell is 7 ft in diameter by 13 ft high. The module cells are made of galvanized, corrugated steel cylinders with welded steel base plates. The cells are encased in concrete for shielding and have a 3-ft-thick concrete cap with a seal.

Each cell is equipped with a drain on the base plate to allow collection of washdown liquid into the facility sump. The sump is equipped with a high-level alarm and must be manually set up for transfer of fluid. The sump pump discharge is capped, and sump water analysis is to be done before any uncapping and transfer of liquid. The sump and sump pump are contained in a concrete structure having two massive, gasketed covers--one 3-ft-thick concrete and one 1-ft-thick concrete. The concrete cells, their covers, and the sump and its covers create a barrier to release of radionuclides.

10.5.1.3 Potential Impacts of Floods

The Three Mile Island Nuclear Station site is protected by a rock-armored levee system that completely surrounds the station structures. The levees range in elevation from 304 ft MSL at the southern (downstream) end to about 310 ft MSL at the northern (upstream) end. The levees were designed and constructed to protect the plant from a flood having a peak discharge or runoff rate of 1,100,000 cubic feet per second (cfs). The height includes a margin of safety to allow for settlement, wind wave activity, and larger floods. The design of safety-related plant facilities inside the levee system will accommodate a PMF having a peak discharge rate of about 1,600,000 cfs. The Hurricane Agnes flood of June 1972 had a peak discharge of 1,020,000 cfs (less than the levee design flood). Although portions of the plant site were flooded during Agnes, construction of the levee system had not been completed at that time, and flood waters inundated portions of the island through unfinished sections of the levee. Had the levees been in place, no flooding would have occurred inside the levee system, as indicated in Table 10.17, in which levee and flood elevations at Three Mile Island are summarized.

Table 10.17. Summary of Levee and Flood Elevations at Three Mile Island

	Elevations (ft MSL)				
Location	Levee Elevation	1936 Flood Level[a]	Agnes Flood Level[b]	Design Flood Level[c]	Probable Maximum Flood Level[d]
South end of levee	304	296.4	299.4	303.0	308.5
Unit 2 intake	305	296.5	300.4	303.5	309.2
North end of levee	310	298.0	301.3	304.5	309.8

[a] Peak discharge of 740,000 cfs--flood of record prior to 1972.

[b] Peak discharge of 1,020,000 cfs--flood of record.

[c] Peak discharge of 1,100,000 cfs--minimum freeboard is about one foot at south end.

[d] Peak discharge of 1,600,000 cfs--levee is overtopped at downstream end.

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The temporary radwaste storage facility was designed to provide shielded storage for 28 spent EPICOR I and II liners (16 cells 4.5 ft in diameter by 8 ft high, and 12 cells 7 ft in diameter by 8 ft high). The cells are galvanized, corrugated metal cylinders that have steel plates welded to one end to provide a bottom. The cells are placed on compacted earth fill in the Unit 2 desilting basin and backfilled with compacted earth. Each cell is provided with a 16-ton concrete shield cover (Appendix D). The elevation of the top of the cells is below the flood control dike elevation of 304 ft Maximum Sea Level (MSL) at the downstream side of the island. Any flood level exceeding the elevation of the dike would result in water levels greater than the top of the cells, but the cells are sealed against leakage by the welded base plate and the massive, sealed concrete top.

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Each cell is equipped with a drain on the base plate to allow collection of washdown liquid into the facility sump. The sump is equipped with a high-level alarm and must be manually set up for transfer of fluid. The sump pump discharge is capped, and sump water analysis is to be done before any uncapping and transfer of liquid. The sump and sump pump are contained in a concrete structure having two massive, gasketed covers--one 3-ft-thick concrete and one 1-ft-thick concrete. The concrete cells, their covers, and the sump and its covers create a barrier to release of radionuclides.

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The Three Mile Island Nuclear Station site is protected by a rock-armored levee system that completely surrounds the station structures. The levees range in elevation from 304 ft MSL at the southern (downstream) end to about 310 ft MSL at the northern (upstream) end. The levees were designed and constructed to protect the plant from a flood having a peak discharge or runoff rate of 1,100,000 cubic feet per second (cfs). The height includes a margin of safety to allow for settlement, wind wave activity, and larger floods. The design of safety-related plant facilities inside the levee system will accommodate a PMF having a peak discharge rate of about 1,600,000 cfs. The Hurricane Agnes flood of June 1972 had a peak discharge of 1,020,000 cfs (less than the levee design flood). Although portions of the plant site were flooded during Agnes, construction of the levee system had not been completed at that time, and flood waters inundated portions of the island through unfinished sections of the levee. Had the levees been in place, no flooding would have occurred inside the levee system, as indicated in Table 10.17, in which levee and flood elevations at Three Mile Island are summarized.

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[a] Peak discharge of 740,000 cfs--flood of record prior to 1972.

[b] Peak discharge of 1,020,000 cfs--flood of record.

[c] Peak discharge of 1,100,000 cfs--minimum freeboard is about one foot at south end.

[d] Peak discharge of 1,600,000 cfs--levee is overtopped at downstream end.

The PMF is derived in a very conservative manner and represents the upper limit of flood potential at a given location, such that there is virtually no chance of the PMF being exceeded. Based on flow frequency data developed by the NRC staff, the Agnes flood of 1972 has a probability of occurrence of less than once in 1000 years; the design flood has a recurrence interval of less than once in 2000 years; and the PMF has an undefined probability, but the occurrence of such a flood is less likely than the design flood. However, if the worst-case flood (PMF) were to occur, the flood level would be about 308 ft MSL. In this case, the top of interim storage facility cells and the concrete storage facility walls, having respective elevations of 304.0 and 305.33 ft MSL, would be flooded.

The prefilter material, resins, and evaporator bottoms are contained, however, in sealed steel containers, which are stored in the cells of the storage facilities. The cells provide a second barrier to release of radioactive material. In order for radioactive waste to escape, both barriers would have to develop an undetected leak path coincident with the PMF.

In view of the very low likelihood of the occurrence and short duration of any flood that would overtop the protective dikes and the low likelihood of simultaneous breaching of the double containment afforded by the storage facilities, it is considered highly unlikely that flooding could cause release of radioactive material.

10.5.2 Potential Releases Due to Tornado

The principal factors that must be considered in evaluating the potential impact of a tornado are the high winds, the pressure drop in the center of a tornado, and the danger of tornado-generated missiles. The design basis tornado for the TMI-2 site has wind velocities up to 360 mph, a pressure drop of 3 psi in 3 seconds, and potential tornado missiles as given in Table 10.18. The facilities and procedures at TMI-2 have been reviewed with these potential factors in mind, and have been found to be adequate to protect the environment and the health and safety of the public from potential effects of tornados.

Table 10.18. Significant Design Basis
Tornado Missiles

Missile	Weight (lb)	Impact (ft^2)	Impact Velocity (mph)
Utility pole	1200	1.0	200
Passenger auto	4000	30	100
Concrete fragment	4500	30	60
Wood plank	110	0.33	360

10.5.2.1 Tornado Probabilities

The probability of a particular point being affected by a tornado is a function of the number of tornados occurring, on the average, in the vicinity and the average path area affected by each. For a 40-year period of record, an average of one tornado per year was reported for the two-degree square (8000 square miles) containing the TMI-2 site. From 1955 through 1967 an average of one tornado per year in the one-degree square containing the site (4000 square miles) was reported. This higher apparent frequency is due to more complete observation and reporting and is used for these evaluations.

Typical tornados are a quarter-mile in diameter. The average path length of tornados in the vicinity of the site is 8.5 miles, but for conservatism 10 miles is used, giving a path area of 2.5 square miles. The probability of a tornado affecting the TMI-2 site is therefore 6.3×10^{-4} per year, or once in 1600 years.

The potential for release of radioactive material is not just a function of the probability of any tornado striking the site, but must also include consideration of severity. The design basis tornado is a very severe one, and one that is not likely to occur in Pennsylvania. Based on observations mainly in the Midwest and South Central United States, the combined probability of a tornado having maximum winds above 275 mph striking the site is 10^{-7} per year. The design basis tornado, with wind speeds of 360 mph, would have a significantly lower probability of occurance.

10.5.2.2 Potential For Water-Processing-Related Releases

The procedures used and protection provided during water processing preclude tornado-induced releases that could endanger the environment or public health and safety. The protection described in Section 10.5.2 is sufficient to mitigate the consequences of high tornado winds, pressure drop, and missiles. Tornado missiles cannot penetrate the safety-related structures (reactor building and the auxiliary and fuel handling buildings) and cannot compromise the integrity of the casks housing the EPICOR II prefilters and liners in the chemical cleaning building.

The exposed processed water storage tanks are susceptible to tornado damage from high winds or missiles. These tanks could release processed water containing an average of less than 0.6 μCi/mL tritium. This tritiated water would likely be drawn into the tornado and then be dispersed by the violent mixing action of the air flow inside the tornado. Even if all the available processed water (about 1.5×10^6 gallons) were removed from tanks by the tornado and blown into the Susquehanna River, and if the river were at its 50 percentile flow rate of 20,000 cfs (9×10^6 gpm), maximum permissible concentration (MPC) limits (10 CFR Part 20) for tritium are not likely to be exceeded.

The York Haven Dam and the Red Hill Dam form a 10,000 acre-foot (3×10^9 gallon) water impoundment in the vicinity of the TMI-2 site. Based on downstream channel volume, it is assumed conservatively that 20 percent of the water impounded is effective in diluting the 1.5×10^6 gallons of processed water, resulting in a downstream tritium concentration of 2.5×10^{-3} μCi/mL, which is 83 percent of the MPC limit for continuous release of tritium (3×10^{-3} μCi/mL). Since all users of surface water downstream of Three Mile Island are also downstream of York Haven Dam, the sudden, one-time release of processed water would pose a very minimal risk to the environment.

10.5.2.3 Potential For Releases From Waste Storage Areas

Both the interim waste storage facility and the concrete waste storage facility are fully protected from the effects of tornados. The cells of both facilities are covered with 3-ft-thick concrete tops and both are essentially at grade elevation. Tornado winds, pressure drop, and missiles cannot compromise the integrity of the cells.

10.5.3 Potential for Releases Due to Aircraft Impact

The potential risk to public health and the environment has been evaluated for the impact of large aircraft on liquid and solid radioactive waste processing and storage facilities. Critical safety-related structures (Category 1) on the site, such as the reactor building and the auxiliary and fuel handling buildings, originally were designed to withstand the impact of large aircraft (300,000 pounds) at 200 knots. Therefore, the sources of relatively high-level radioactivity, such as the nuclear fuel and the reactor building sump water, are protected from the effects of aircraft impact. Likewise, any cleanup operations, that may occur inside these buildings also are protected.

There are, however, some sources of radioactivity outside of these buildings designed to withstand aircraft impact, and the potential vulnerability of these sources has been evaluated. The facilities or components that have been considered are the chemical cleaning building housing EPICOR II, the concrete waste storage facility, the interim waste storage facility, the borated water storage tank (which is used to store processed water containing tritium), and the processed water storage tanks.

10.5.3.1 Possibility of Aircraft Impact

An evaluation has been done by the staff to determine the probability of impact of large aircraft on radioactive material storage facilites at TMI-2. On page 31 of Reference 7, the staff determined the impact probability for each potential target to be 1.6×10^{-8} per year. Therefore, the aircraft impact probability is a factor of approximately 6 smaller than the 10^{-7}/yr probability that is often used as a limit beyond which events are considered highly unlikely or incredible, and not seriously considered in design.

10.5.3.2 Potential For Water-Processing-Related Releases

During processing of liquids, contaminated water is not stored in tanks susceptible to aircraft impact. It is either pumped from a protected building directly to a water treatment system, such as EPICOR II, or it is decontaminated inside a protected building, as would be the case with the submerged demineralizer system.

Processed Water Storage

The processed water resulting from cleanup operations contains tritium and may be stored outside in tanks vulnerable to aircraft impact, such as the borated water storage tank or the processed water storage tanks. However, the potential risk associated with this storage is minimal because of the very low probability of aircraft impact (on the order of 4×10^{-10}) and the low consequence of release of the tritiated water. As was described in Section 10.5.1, there is a flood protection dike around the site, and this dike would serve to contain processed water that might be released by aircraft impact. The expected average tritium concentration in processed water after cleanup of reactor building sump water and reactor coolant system water is less than 0.6 µCi/mL, and the slow leakage or evaporation of this processed water does not present a public health hazard.

Chemical Cleaning Building (EPICOR II)

The chemical cleaning building tanks used for interim storage of processed water are susceptible to failure caused by an aircraft impact, but the water would be contained by the flood protection dike. The highly radioactive EPICOR II prefilters and liners are protected by 12-inch-thick reinforced-concrete walls surrounding the processing area. The prefilter/demineralizer is installed inside a 12-inch-thick cylindrical concrete cask, and the cask is surrounded by a lead brick wall 5 inches thick. The top of the prefilter/demineralizer is covered with a lead shield and 5-inch-thick steel lid. The second- and third-stage demineralizers are similarly protected. The room containing the demineralizer stages is further protected by 24-inch-thick concrete walls inside the 12-inch-thick exterior walls. Considering the substantial amount of impact resistance and the very low probability of aircraft impact, it is considered highly unlikely that radioactive releases would occur from the chemical cleaning building as a result of aircraft impact.

10.5.3.3 Potential for Releases from Waste Storage Areas

Relatively high-level wastes such as water processing prefilters and liners will be stored in the interim radwaste storage facility on the site. The potential for releases occurring as a result of aircraft impact on these waste storage areas was evaluated as being very small. The temporary radwaste storage facility, described in Section 10.5.1.1, had prefilter and resin material protected from aircraft impact by the 3-ft-thick concrete cell top and the primary steel container. These have been removed to the interim radwaste storage facility. The interim radwaste storage facility described in Section 10.5.1.2 also has a 3-ft-thick concrete cell top and is well protected from aircraft impact. In view of the substantial concrete protection afforded the waste and the very low probability of aircraft impact, it is highly unlikely that such an accident would result in any release that would endanger public health or safety.

10.6 PSYCHOLOGICAL-SOCIOECONOMIC EFFECTS

The staff concludes that low-level chronic distress could continue throughout the cleanup period, and some groups will be more affected by stress than others. Most of the impacted public currently demonstrate no psychological effects that have detrimental health consequences. Exact estimates of long-term psychological consequences associated with the accident or the cleanup are not available. However, one study suggests that even among the most vulnerable of the impacted groups, the distress levels measured on the first anniversary of the TMI accident are below the threshold associated with health problems and are not much higher than control group scores.

The staff believes that the potential level of distress generated by the decontamination process could be related to the present psychological setting and factors surrounding decontamination. The present psychological setting for those who are distressed is characterized by uncertainty over the effect of released radiation on health and future generations, NRC's and Met-Ed's lack of credibility, and the believed inability of Met-Ed to safely conduct and NRC to review the various decontamination procedures. Those factors which could contribute to the depth and breadth of future distress levels include the length of time required for overall decontamination and for specific decontamination procedures. In general, it is the staff's belief that as time spent on decontamination activities increases, the consequences of distress could increase. Other factors that could influence future levels of potential distress include the believed probability of accidents, the type and duration of media coverage, the credibility of those managing the decontamination process, and believed levels of safety and health impacts.

Considerable public concern has been expressed regarding the prospect of processed water being released into the Susquehanna River. Much of this concern centers on possible disruption of recreational and commercial fishing activity on the Bay due to a perceived threat to health, whether justified or not. The staff concluded (Section 7.2.5.4) that the disposal of tritiated water in the Susquehanna has the potential to produce dispersed socioeconomic impacts as a consequence of distress and uncertainty. These impacts could include the temporary avoidance of drinking water, of recreational opportunities, and of seafood and waterfowl supplied by the river or the Bay. Potential economic losses to watermen, processors, restaurants, and other retail and service outlets might be significant. As discussed in Section 7.2.5.5, the State of Maryland will be undertaking a study of the potential economic losses to persons who depend upon Chesapeake Bay activities. There is also the potential for public uncertainty over radiation effects and widespread economic loss. This condition could require mitigative measures to help reduce public anxiety if the Susquehanna disposal option is selected. An appropriate mitigation program should seek to relieve several problems including (1) the NRC's and Met-Ed's lack of credibility, (2) the limited knowledge reflected in some of the media and general public perceptions of radiation-related impacts, (3) the inadequate dissemination of consistent information by public agencies, and (4) the inadequate incorporation of socioeconomic and ecological considerations in the decision on the timing of disposal.

The transportation of waste was also indicated as having the potential for producing socioeconomic impacts. The expected range of number of truck shipments is 353 to 997. These shipments would be made over a period of 5 to 9 years; hence, the average rate of shipments during this period would range from about one truckload every three days to about one truckload every five days. Those living along or in proximity to the truck route could be subjected to stress. The staff judgment is that the marketability of residential property abutting the route through Middletown could be adversely impacted during the period of shipment. Also, the staff would expect that some residents will alter their daily schedules to avoid the shipping route at certain times. The staff believes that the potential increase in stress and annoyance can be mitigated through a process that considers the apprehensions of those living in the Middletown transportation corridor. To accomplish this end, the staff considers it necessary to continue the ongoing public dialogue to discuss the structural integrity of shipping containers, the low probability of radiation leakage under accident conditions, and the optimal timing for shipping waste.

Although they would have negligible offsite effects, airborne releases as a result of postulated accidents could create public uncertainty over health and safety with consequent impacts to the agricultural sector. As noted in Section 3.6.2.3, "milk juggers"--local dairy farmers who sell milk directly to retail customers--experienced sales declines subsequent to the TMI accident. The staff believes that because of the small quantity of radioactive material potentially involved and the existing monitoring programs, such losses, should they occur, would be of short duration.

10.7 IRREVERSIBLE AND IRRETRIEVABLE COMMITMENTS OF RESOURCES

Many resources will be used in the cleanup of TMI Unit 2. Some of those uses involve irreversible and irretrievable commitments. Irreversible commitments concern changes set in motion by the proposed action that at some later time could not be altered so as to restore the present order of environmental resources. Irretrievable commitments are the use or consumption of resources that are neither renewable nor recoverable for subsequent utilization.

10.7.1 Commitments Considered

The types of resources of concern in this case can be identified as: (1) material resources--materials of construction, renewable resource material, and depletable resources consumed, and (2) nonmaterial resources--including a range of beneficial uses of the environment.

Resources that generally may be irreversibly committed by the construction and cleanup are: (1) construction materials that cannot be recovered and recycled with present technology, (2) materials that are rendered radioactive but cannot be decontaminated, (3) materials consumed or reduced to unrecoverable forms of waste, (4) the atmosphere and water bodies for disposal of waste effluents, to the extent that other beneficial uses are curtailed, and (5) land areas rendered unfit for their original uses.

10.7.2 Material Resources

10.7.2.1 Construction Materials

Materials of construction are almost entirely of the depletable category of resources. Concrete and steel constitute the bulk of these materials. No commitments have been made on whether these materials will be recycled when their present use terminates. The amounts to be used for the cleanup are not known at this time because detailed procedures have not been determined; however, quantities of construction materials used will be very small compared to annual U.S. production of these materials.

10.7.2.2 Replaceable Components and Consumable Materials

Materials consumed or reduced to unrecoverable status are chemicals such as detergents used for the cleanup and gasoline and diesel fuel used in vehicles bringing materials and the workforce to and from the site, for haulage onsite, and for transportation of waste material offsite. The amounts used in local transport are unpredictable; however, the transportation of wastes to licensed disposal facilities will require from 1/2 to 1 1/2 million gallons of diesel fuel, depending on the distance to the burial sites and the waste forms selected.

The materials used for processing, immobilizing, and packaging the wastes have not yet been chosen, but the candidates are ion-exchange resins, vinyl ester styrene, portland cement, and bitumen. The organic material, resins, vinyl ester styrene, and bitumen are derived from petroleum, but the quantities required are small in comparison to the total world production of petroleum. Portland cement may be used for immobilizing the wastes or as an alternative to processing the wastes. Here again, the quantities expected to be used are small compared to U.S. production.

10.7.3 Water and Air Resources

Because of the rapidly renewable nature of the Susquehanna River and the regenerative powers and vast dispersive capacity of the atmosphere, the use of these resources to dilute and disperse the effluents of chemicals and radioactive materials from the cleanup of TMI-2 is not considered to represent irreversible or irretrievable commitments of these resources.

10.7.4 Land Resources

Land required for the burial of low-level radioactive wastes is estimated to range from 1/2 to 1 acre, depending upon the amount and volume of the wastes. High activity wastes could be accommodated within these estimates, assuming land burial is permitted. Nuclear fuel from the reactor would require from 66 ft^2 to 216 ft^2 in a storage pool at a presently undesignated location.

10.8 SUMMARY OF ECONOMIC COSTS OF THE TMI-2 CLEANUP

Numerous alternatives have been presented in each of the previous sections of this statement. Because each section of this statement represents a particular evolution in the cleanup of TMI, the combinations of alternatives and evolutions produce an enormous number of reasonable paths that could be selected for the actual cleanup. Therefore, the staff has developed bounding cost estimates for each of the major sections of this statement. Table 10.19 summarizes these costs.

The costs stated in Table 10.19 have been developed based upon constant 1980 dollars and do not include the time value of money nor the inflationary conditions that may exist in the future. Additionally, costs have been excluded unless they can be directly associated with a given alternative. Costs that have been excluded from Table 10.19 are: maintaining the plant in a safe shutdown condition, site security costs, licensing costs, housekeeping costs, receipt/ inspection costs, planning and scheduling costs, warehousing costs, material handling costs, general technical surveillance, including quality assurance, health physics and engineering. In addition, costs have not been included for construction of general support facilities, such as office and general laboratory buildings.

Thus, the economic data presented here, and elsewhere in the PEIS, are only for the purpose of comparison among alternatives of direct costs. The costs given here can not be summed to give a total cost estimate for the cleanup. Some generalities can, however, be stated:

- The selection of alternatives will be an important factor in determining the total cleanup cost, but the timing of the cleanup will most likely dominate the total cost.
- There is a large uncertainty in the relative costs for most of the alternatives. The uncertainty in individual cost estimates could, in some cases, be greater than the cost differences between alternatives.

Because of the uncertainty in estimated costs among the alternatives, and because public and worker safety, as well as protection of the environment, are the paramount concerns, review of alternatives based on costs is a secondary consideration.

Table 10.19. Summary of Alternative Economic Costs for the Cleanup of TMI-2[a] (thousands of dollars)

PEIS Section		Estimated Cost Range	
		Low	High
5.1	AFHB cleanup	$16,000	$ 22,000
5.2	Reactor building cleanup	25,000	63,000
6.2	Cooling system inspection	1,400	5,900
6.3	RPVH and internals removal	3,600	6,600
6.4	Core examination and defueling	12,000	16,000
6.5	RCS decontamination	2,600	6,900
7.1	Liquid waste treatment	21,000	29,000
7.2	Disposal of processed accident water	100	11,000
8.1	Process solid wastes	23,000	27,000
8.2	Chemical decontamination solutions	2,000	10,000
8.3	Solid materials	700	7,100
9.0	Storage, transportation, and disposal of solid waste	2,600	11,000

[a] This summary of economic costs is for the purpose of making comparisons among alternatives. The relative costs can not be summed to arrive at a total cost estimate for the entire cleanup.

References--Section 10

1. "Ionizing Radiation: Levels and Effects," Volume II: Effects, a Report of the United Nations Scientific Committee on the Effects of Atomic Radiation to the General Assembly, 1972.
2. "Effects on Populations of Exposure to Low Levels of Ionizing Radiation," National Academy of Sciences Advisory Committee on the Biological Effects of Ionizing Radiations, November 1972.
3. NRC Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Office of Standards Development, U.S. Nuclear Regulatory Commission, October 1977, Rev. 1.

4. "Age-Specific Radiation Dose Commitment Factors for a One-Year Chronic Intake," Battelle Pacific Northwest Laboratory-NRC, NUREG-0172, November 1977.

5. "Environmental Survey of Transportation of Radioactive Material to and from Nuclear Power Plants." U.S. Nuclear Regulatory Commission, WASH 1238, December 1972, and Supplement, NUREG-75/038, April, 1975.

6. L. Garfinkel and E. Silverberg, "Cancer Statistics, 1979," American Cancer Society Professional Education Publication, p. 7, 1979.

7. L.J. Chandler and S.A. Treby, "NRC Staff Posthearing Memorandum Regarding Aircraft Crash Probability Issue," Before the ALAB in the Matter of Metropolitan Edison Company, et al. (Three Mile Island Nuclear Station, Unit 2). Docket No. 50-320, April 30, 1980.

11. ENVIRONMENTAL RADIOLOGICAL MONITORING

11.1 INTRODUCTION

The radiological environmental monitoring around the TMI site and nearby communities during the decontamination of Unit 2 would be performed by (1) Metropolitan Edison Company (the licensee), (2) the U.S. Environmental Protection Agency (EPA), (3) The Commonwealth of Pennsylvania, (4) the U.S. Department of Energy (DOE), (5) the Nuclear Regulatory Commission (NRC), and (6) the State of Maryland. Each program is summarized in the following subparagraphs; a more complete description is given in the EPA report, "Long-Term Environmental Radiation Surveillance Plan for Three Mile Island," 1981, which is provided as Appendix M to this statement.

11.2 METROPOLITAN EDISON COMPANY RADIOLOGICAL MONITORING PROGRAM

The Met-Ed radiological environmental monitoring program which will be in effect during the decontamination of Unit 2 is a combination of the TMI-1 and TMI-2 Environmental Technical Specification required programs and increased monitoring activities which were initiated after March 28, 1979. This monitoring program is subject to change based upon review of the results and any requests for additional monitoring.

The licensee's radiological monitoring program is comprehensive, covering sampling of air, milk, water, fish, aquatic plants, sediments, miscellaneous food products, and exposure rates in the environs in and around the TMI facility to a distance of about 21 miles.

The licensee's air sampler network consists of eight stations which are sampled weekly using both air particulate filters and charcoal cartridges. Air particulate samples are analyzed weekly for gross beta activity, and gamma spectral analysis is also performed monthly. A radioiodine analysis is performed on the charcoal cartridge. For a quarterly composite of the air particulate samples, analyses for Sr-89 and Sr-90, gross alphas, and gamma spectra are made.

Met-Ed's milk network samples semimonthly from five farms in the offsite area. Radioiodine and gamma spectra determinations are performed on these samples. A Sr-89, Sr-90 analysis is performed on quarterly composites of these samples.

Water samples from Met-Ed's offsite water sampling network are collected from eight stations. These samples are composited hourly over a two-week period utilizing automatic water samplers. These semimonthly samples are analyzed for iodine (semimonthly), gamma scan and gross beta analyses on monthly composite, tritium on a monthly and quarterly composite, and Sr-89 and Sr-90 on a quarterly composite. In addition, grab samples are taken weekly at two surface water stations. These are composited and the above analyses are performed. Daily grab samples are also taken from the plant discharge and composited for the above analyses.

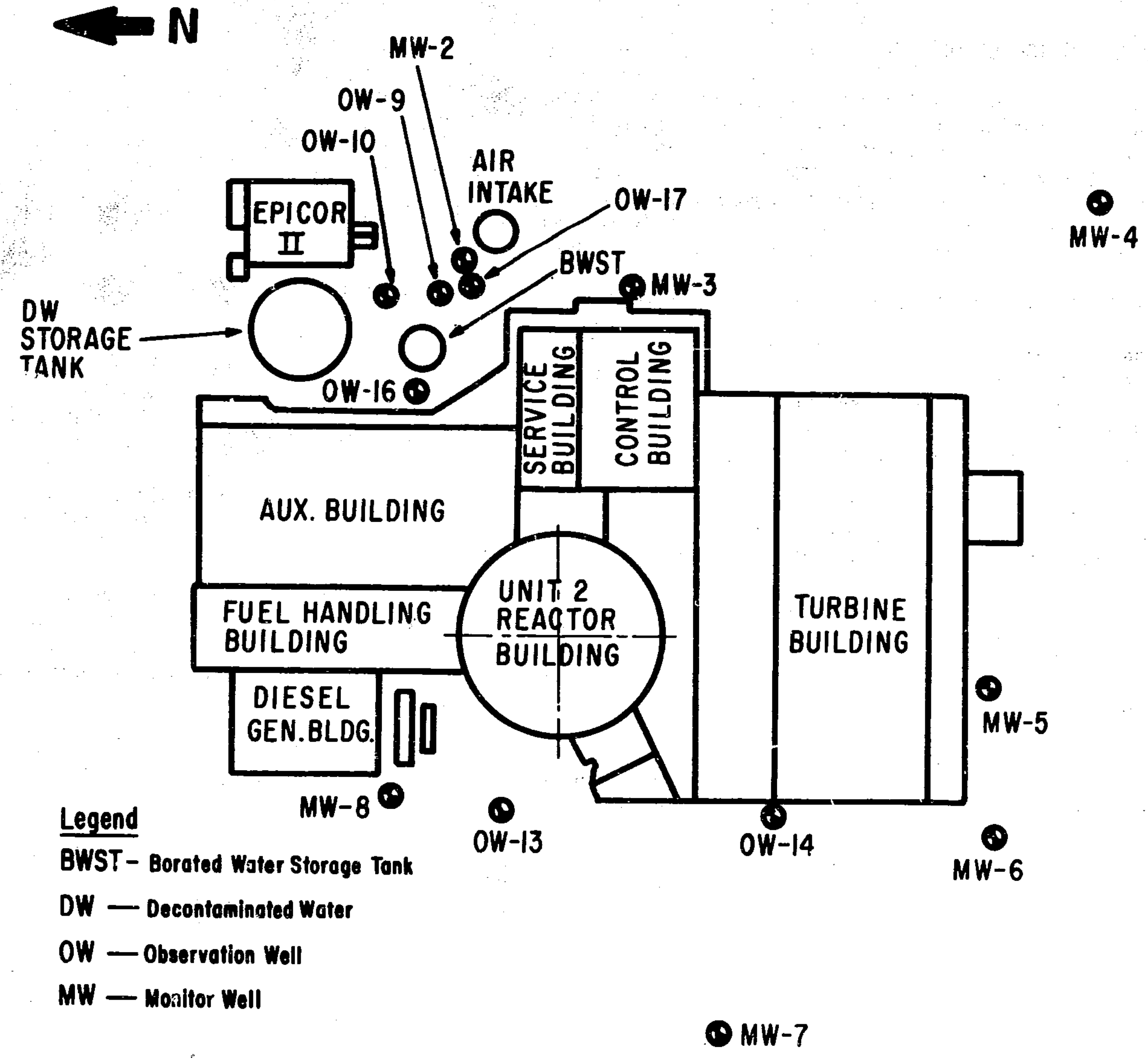
Fish, aquatic plants, and aquatic sediments are sampled periodically, as well as miscellaneous food products as they become available.

Met-Ed operates an extensive thermolumenescent dosimeter (TLD) network for monitoring environmental exposure rates in the area around the site. These dosimeters are exchanged on a monthly (20 locations) and a quarterly (53 stations) basis.

Met-Ed has a groundwater monitoring program (see Fig. 11.1) that presently samples from 15 observation and monitoring wells. Tritium analysis and gamma scans are performed on the samples taken.

11.3 U.S. ENVIRONMENTAL PROTECTION AGENCY RADIOLOGICAL MONITORING PROGRAM

EPA has been designated by the Executive Office of the President as the lead Federal agency for conducting a comprehensive, long-term environmental radiation surveillance program as a followup to the accident at TMI-2. As of December 31, 1980, EPA operates a network of 13 continuous


N
MW-2
OW-9
OW-10
AIR INTAKE
OW-17
EPICOR II
BWST
MW-3
MW-4
DW STORAGE TANK
OW-16
SERVICE BUILDING
CONTROL BUILDING
AUX. BUILDING
UNIT 2 REACTOR BUILDING
FUEL HANDLING BUILDING
TURBINE BUILDING
DIESEL GEN.BLDG.
MW-5
MW-8
OW-13
OW-14
MW-6
Legend
BWST - Borated Water Storage Tank
DW — Decontaminated Water
OW — Observation Well
MW — Monitor Well
MW-7

COMMENTS:

1. MW-1 LOCATED IN THE NORTH PARKING LOT AT COORDINATES N 301,460.04 E 2,286,538.94

2. OW-15 LOCATED ON SOUTH END OF ISLAND AT COORDINATES N 292,985.44 E 2,287,765.09

Figure 11.1. Monitor Well Locations at TMI-2.

air-monitoring stations at radial distances ranging from 0.5 mile to 5 miles from TMI. Five miles was established as the point well beyond that which EPA expects to detect any emissions from TMI-2. Each station includes an air sampler, a gamma rate recorder, and three TLDs. The air sampler units operate at approximately 2 cfm, and particulate samples are collected from each station and analyzed typically three times per week. Charcoal filters are collected and analyzed weekly. All samples are analyzed by gamma spectroscopy at EPA's TMI Field Station, Middletown, using a Ge(Li) detector with a lower limit of detection for Cs-137 or I-131 of about 25 pCi (0.15 pCi/m^3 for a 48-hour sample).

Each monitoring station is equipped with a gamma-rate recorder for measuring and recording external exposure. Recorder charts are read on the same schedule used for air sample collection, and the charts are removed weekly for review and storage at EPA's laboratory in Las Vegas, Nevada. EPA is presently planning to install telemetered gamma monitors at these stations. When this change is effected, EPA will revise its collection and analysis of charcoal and particulate filters to once a week.

Thermoluminescent dosimeters have been placed at each monitoring station as well as at a representative number of population centers surrounding TMI. Locations are shown in Appendix M of this programmatic statement (Appendix D of EPA surveillance plan). These dosimeters are read quarterly.

In addition to the above, weekly compressed gas samples are taken at Yorkhaven, Goldsboro, Middletown, and at the TMI Observation Center opposite the plant and sent to EPA Las Vegas for a determination of krypton-84. Tritium in air samples will also be collected from these stations.

The EPA's base long-term program discussed above will continue and can be augmented in the future if particular decontaminations make it necessary. Such augmentation of the monitoring program for the venting of the Kr-85 from the reactor building consists of survey meter and ion chamber measurements, collection of compressed air samples for Kr-85 analysis, and intensified collection of samples from routine air monitoring stations.

EPA also collects and analyzes water samples as follows: (EPA does gamma spectroscopy, DER analyzes for tritium, gross alpha and gross beta: weekly composites are analyzed for strontium-89 and 90 at the Eastern Environmental Radiation Facility, EPA, Montgomery, Ala.)

(1) TMI outfall (all plant discharge, both units) - daily,

(2) Lancaster Water Works intake - daily,

(3) City Island - (upstream river water) - weekly, and

(4) Sediment pond, TMI (runoff water) behind Unit 2 cooling tower.

There is a continuous gamma monitor on the 001 TMI outfall with a high-level alarm that automatically alerts EPA and DER to the presence of gamma activity in the water in excess of 1000 pCi/L Cs-137 (1/20 of permissible level).

EPA reports all results of monitoring measurements from their baseline program on Friday each week to the public and news media.

11.4 COMMONWEALTH OF PENNSYLVANIA RADIOLOGICAL MONITORING PROGRAM

The Department of Environmental Resources of the Commonwealth of Pennsylvania operates three continuous air-sampling stations; one at the Evangelical Press Building in Harrisburg, one at the TMI Observation Building, and one in Goldsboro near the boat dock. Each air-sampling station consists of a particulate filter followed by a charcoal cartridge. The filters and cartridges are changed weekly; the particulate air samples are gamma scanned and beta counted for reactor-related radionuclides. The particulate air samples are composited quarterly and analyzed for Sr-89 and Sr-90. The charcoal samples are gamma scanned for reactor-related radionuclides. The Commonwealth, however, does not have the capability to sample or analyze for Kr-85.

The Commonwealth's milk sampling has reverted to its routine surveillance program, which consists of monthly milk sampling at two dairy farms near the site. The milk samples will be gamma scanned for all reactor-related gamma-emitting radionuclides. The Commonwealth has TLDs at ten locations which are collected and read monthly. The Commonwealth also will collect local produce and fish in season. The produce and fish samples will be analyzed by gamma spectroscopy for any reactor-related radionuclides. The Commonwealth also participates with EPA in monitoring the principal aqueous outfalls of TMI-2.

11.5 U.S. DEPARTMENT OF ENERGY RADIOLOGICAL MONITORING PROGRAM

11.5.1 Radiological Monitoring Program

DOE will provide soil and vegetation analyses from seven sites semiannually. In-situ gamma spectrometry analyses will be conducted at these seven plus one additional site. TLDs are also in place at these sites plus four state monitoring locations. If levels of radionuclides demonstrate any increase above background levels, the samples will be subjected to detailed radiochemical analyses.

11.5.2 Atmospheric Release Advisory Capacity

DOE has available for significant radiological releases to the atmosphere its Atmospheric Release Advisory Capacity (ARAC). This ARAC system provides independent predictions of the dispersion patterns for such releases based on local meteorological data and National Weather Service reports. These predictions use atmospheric dispersion models which have been verified during many years of field experience and tests in government programs. The predicted dispersion patterns would be provided to EPA, the utility, and NRC to serve as a basis for their positioning of ground-level monitoring teams.

11.5.3 Community Monitoring Program

DOE and the Commonwealth of Pennsylvania are sponsoring a community radiation monitoring program. This program has as its purpose to: (a) provide independent verification of radiation levels in the TMI area by trained local community people, and (b) increase public understanding of radiation and its effects. The approach to achieving this purpose has involved the selection of individuals by local officials from the following 12 communities within about five miles around TMI:

East Manchester Twp.
Londonberry Twp.
York Haven
Lower Swatara Twp.
Conoy Twp.
Goldsboro
Fairview Twp.
Royalton
West Donegal Twp.
Middletown
Newberry Twp.
Elizabethtown

About 50 individuals participated in training classes conducted by members of the Nuclear Engineering Department of the Pennsylvania State University. About 15 training sessions were conducted involving classroom instructions, laboratory training, and actual radiation monitoring in the field. The teams utilized EPA gamma rate recording devices which are in place around TMI and will be supplemented by gamma/beta-sensitive devices which are being furnished by DOE through EG&E Idaho, Inc.

The training sessions provided basic information on radiation, its effects and detection techniques, and included hands-on experience with monitoring equipment in the field. Following the completion of training in the third week of April, team representatives in each of the 12 selected areas began data acquisition from the gamma and gamma/beta-sensitive instruments on a routine basis. Detailed procedures were developed to consolidate the information being obtained into a central point of contact in the Commonwealth of Pennsylvania for dissemination to the press, local officials, and other interested parties on a routine basis. Maintenance and calibration procedures also were developed and were in effect prior to the initiation of routine field monitoring. The community monitoring program was initiated on May 21, 1980, and the results of measurements from this program were reported daily to the public. Presently only the units at Fairview, West Donegal and Newberry continue to be active.

11.6 U.S. NUCLEAR REGULATORY COMMISSION RADIOLOGICAL MONITORING PROGRAM

The NRC operates one air-sampling station located in the middle of the reactor complex. The air sample is changed weekly and will be analyzed by gamma spectrometry. The NRC places two sets of TLDs at 59 locations in the area around TMI-2. Each set contains two lithium borate and two calcium sulfate phosphors. The lithium borate phosphor has the ability to detect beta radiation from Kr-85. Both sets will be read on a monthly basis; however, flexibility exists to read one set at more frequent intervals should conditions warrant.

11.7 STATE OF MARYLAND RADIOLOGICAL MONITORING PROGRAM

In early April 1979, the State of Maryland, Department of National Resources (Radiation Section, Division of Environmental Chemistry) began monitoring the levels of radionuclide activity in fishes and sediments of the Susquehanna River at Falmouth (about 1 mile downstream of Three Mile Island), at several locations near and downstream of Safe Harbor Dam and Holtwood Dam, and in the Conowingo Pond. The Chesapeake Bay also has been monitored at several locations from the mouth of the Susquehanna River to about Worton Point (south of the Sassafrus River). Fishes, oysters, blue crab, and bay sediments have been studied. Monitoring of both the river and bay will continue during the cleanup of Three Mile Island.

Since the accident, the Maryland Department of Health and Mental Hygiene (Division of Radiation Control, Environmental Health Administration) has been testing Susquehanna River water for radionuclide content using weekly composite samples collected at Conowingo Dam. This sampling will continue. For discharging tritiated water at Three Mile Island (should that be authorized), strong consideration will be given to accelerating the program, as appropriate, to monitor radionuclides in the water. Consideration also will be given to sampling upstream of the Conowingo Dam sampling point in order to differentiate more accurately the radionuclides originating at Three Mile Island from those originating at Peach Bottom. Users of Susquehanna River water within Maryland will be advised, as appropriate, on actions to be taken, should that become necessary.

11.8 CONTINGENCY SURVEILLANCE PROCEDURES

Contingency planning for the protection of the public must address the possibility of unplanned releases of airborne radioactivity to the general environment, as well as liquid releases to the Susquehanna River.

In the event of a release of airborne radioactivity in excess of the licensee's Technical Specifications limits, the EPA On-Site Coordinator will be notified by the NRC. NRC health physics personnel would be supported by radiation monitoring equipment and analytical capabilities, including the NRC Region I mobile laboratory. Additional NRC personnel would be onsite within two hours; the location of the mobile laboratory at the time of the occurrence would dictate its response time. The Senior NRC Site Representative will ensure that the EPA On-Site Coordinator has access to current release data and meteorological information. In addition, the DOE Emergency Coordination Center will be notified by NRC and may be requested to provide aerial measurements and plume tracking. The response time for an aircraft to reach TMI can be expected to be from two to three hours under normal conditions, with a six-hour maximum under virtually any condition.

Air sampling will be a measure of inhalation exposure and potential contamination of milk and food crops. Should a prolonged airborne release occur, supplemental air monitoring stations will be established. Ten air samplers will be kept available by EPA for this purpose. EPA also will have three compressed air samplers available for krypton gas sampling. Apparatus to sample air for subsequent tritium analysis also will be available for prompt use by EPA.

Releases of contaminated water that are above the licensee's permitted level for discharge to the Susquehanna River should not occur. The contingency plan for releases above the licensee's permitted level involves prompt confirmation of the released activity by composite sample analysis, followed by notification of the impact of the release to downstream users. EPA's Region III Office will be responsible for notifying adjoining states.

11.9 REPORTING PROCEDURES

There are two types of data-reporting procedures. The first type is designed to distribute information upon which immediate action might be taken and consists of informal reporting methods; the second procedure is designed to provide a verified data base. In addition, there will be a procedure for reporting information to the news media.

11.9.1 Immediate Reporting Procedures

Each of the monitoring agencies will promptly inform the other monitoring participants of confirmed, positive levels of reactor-related radionuclides through the EPA onsite representative. He will promptly relay the information to the other organizations by telephone or in person to each Federal agency and the Commonwealth of Pennsylvania, followed in either case by written documentation of the event.

Periodic meetings are called by EPA at TMI to discuss proposed and ongoing operations which could impact the offsite agencies and to exchange information.

11.9.2 Reporting Data into the Data Base

All data are reported in the format previously specified by EPA. Data from NRC, DOE, and the Commonwealth of Pennsylvania are submitted to EPA monthly for inclusion in the data base. EPA data also are placed in the data base monthly.

On a monthly basis, EPA will place data obtained from Metropolitan Edison and the Commonwealth of Pennsylvania, as well as relevant data from other organizations, into the data base. EPA then will use computer transfers to transmit monthly updates to the data base to the originating organizations for verification. All data will be verified by the originating organization within 15 days of receipt. Any errors will be referenced by sample number for correction. Periodic updates will be made available to all participants.

11.9.3 Reporting Information to the News Media

The EPA is the lead Federal agency responsible for distribution of environmental data to the news media. Each participant will keep each of the other participants in this plan advised in advance of pending media releases concerning TMI. As appropriate, releases will be coordinated with Metropolitan Edison Company.

11.10 MONITORING OF AQUATIC ECOLOGY AND RECREATIONAL FISHERIES OF THE SUSQUEHANNA RIVER

Appendix B of the operating license for TMI-2 contains Environmental Technical Specifications (ETS) that require Metropolitan Edison Company to monitor station effluents and the Susquehanna River for nonradiological impact from operation. The ETS for Unit 2 were issued by NRC on February 8, 1978, and require three years of operational studies of: (1) physical and chemical characteristics of the York Haven Pond of the Susquehanna River (see Fig. 11.2); (2) thermal and chemical characteristics of the cooling water discharge effluent; (3) general ecology of the pond, including benthic macroinvertebrates, fish eggs and larvae, fish populations, fish impingement, and fish egg and larvae entrainment; and (4) recreational fishery creel survey of fishing effort, catches, and harvests in the pond and other nearby fishing areas. Additionally, the ETS require the licensee to make a prompt report to NRC of any unusual or important events, such as fish kills, near or downstream of the station.

These studies are a continuation of those that have been ongoing since Unit 1 commenced operation in 1974. Results have been presented in a series of annual reports as required by the ETS. Annual reports for the years 1974 through 1978 are complete and constitute a five-year preaccident record of environmental conditions in York Haven Pond. The annual report for the first postaccident year of 1979 is complete. These reports have formed the bases for impact assessments in the 1976 NRC Final Supplement to the Final Environmental Statement on the operation of TMI-2 (NUREG-0112) and assessments presented as testimony during the 1977 environmental hearings in Harrisburg. Following the accident, those annual reports plus other independent data sources were used in assessing the status of the aquatic biota and the recreational fishery of the river near TMI, with results contained in NUREG-0596. The annual reports and the postaccident biological assessment are used as informational sources in this PEIS and are contained in References 1 through 7 of Appendix E.

These nonradiological studies do not directly monitor either environmental radiation levels or their dose effects, but they do provide information on ecological status of the river upstream, near, and downstream of the nuclear station, and in all three channels of the York Haven Pond (see Appendix E, Fig. E.1 for reference). Fish disease, parasite, abnormality, and mortality conditions are routinely monitored during the river studies. These are useful for short-term (mortalities or fish kills) and long-term (disease) effect studies after an accident or radiological release event. The studies have defined (and continue to monitor) those areas of the

Figure 11.2. Sampling of Susquehanna River. Environmental technicians routinely sample water upstream and downstream from TMI. The samples are analyzed for several materials, although no accident-generated water may be discharged from the station. (Official TMI Photo)

river near TMI that are most important to the fishes as spawning and nursery areas. Radiological releases (planned or accidental) therefore can be evaluated in relation to their presence or absence in important areas of the river. Monitoring of the recreational fishery permits an examination of TMI-2 operation on the fishery resource at the point of exploitation. The studies define those areas of the river most important to fishermen and provide information on the food chain that can be used for assessing the liquid radiological pathway to man via finfish consumption. These programs monitor the aquatic biota and sport fishery in that segment of the Susquehanna River where the TMI effluent first enters, where it is the least dilute, and where effects (if any) would be seen first.

The Commonwealth of Pennsylvania National Pollutant Discharge Elimination System (NPDES) Permit requires in-plant monitoring of thermal and chemical effluents, as well as entrainment and impingement. The combination of the studies required by the NPDES and the ETS provide a spectrum of data needs for analysis of the environmental status of the aquatic resources of the river. Both groups of studies are continuing, and data summaries are provided to the respective agencies on a monthly basis. The annual report required by the ETS is on a calendar year basis.

12. CONCLUSIONS

In this programmatic environmental impact statement, the NRC staff has evaluated the environmental impacts and other costs and benefits associated with the proposed cleanup of Three Mile Island Unit 2. As a result of its evaluation, the staff has made the following findings and conclusions:

12.1 Conclusions on Environmental Impacts and Other Costs

1. The cumulative radiation dose received by the entire work force would be in the range of 2000 to 8000 person-rem for the whole cleanup program. It is predicted that less than one additional cancer death attributable to exposure to radiation will occur among the work force (the death rate from cancer due to other causes among the U.S. population averages approximately 200 deaths per 1000 people). Not more than two additional genetic effects in descendents of the workers are expected to occur (among the U.S. population, approximately 60 genetic defects can be expected per 1000 people). This is the most significant radiological impact expected from the cleanup activities. (See Sec. 10.2)

   The occupational dose to an individual worker will be limited to 3 rem/quarter in accordance with 10 CFR Part 20; however, the exact dose below 3 rem/quarter to any one individual cannot be determined due to lack of information about specific work assignments.

2. Throughout the cleanup, any anticipated releases to the environment must be controlled by the licensee in accordance with the staff's proposed effluent criteria to conform to the individual dose design objectives listed in 10 CFR Part 50, Appendix I, as mandatory limits. The total-body dose design objectives are 15 mrem/year from airborne particulate releases and 3 mrem/year from liquid releases. (See Sec. 10.3.4)

   Decontamination methods and technology are available which can be used to complete the cleanup in accordance with the offsite dose limits stated above. If the cleanup is conducted in accordance with the staff's proposed effluent criteria, the staff estimates that, for the entire cleanup, the total body dose to the maximum exposed individual offsite will range from 0.8 to 2.3 mrem for gaseous effluents and from 0.0015 to 1.1 for liquid effluents. The cleanup is expected to take from 5 to 9 years to complete.

3. An individual offsite receiving the maximum estimated dose resulting from atmospheric releases during the entire cleanup (0.8 to 2.3 mrem) would incur an estimated increased risk of dying from cancer of between 1 in 2 million and 1 in 600,000, and an increased risk of a genetic effect to offspring over the next 5 generations of between 1 in 300,000 and 1 in 100,000. As a result of liquid releases which may occur over the entire cleanup period, an individual receiving the maximum estimated dose (0.0015 to 1.1 mrem) would incur an estimated increased risk of dying from cancer of between 1 in 1 billion and 1 in 1 million, and an increased risk of genetic effect to offspring over the next 5 generations of between 1 in 200 million and 1 in 200,000. (See Sec. 10.3)

   If an offsite dose equal to the staff's proposed (10 CFR Part 50, Appendix I) atmospheric annual limit (15 mrem/yr., total body) were received, that individual would incur an estimated increased risk of dying from cancer of 1 in 100,000 and an increased risk of a genetic effect to offspring over the next 5 generations of about 1 in 20,000. An offsite dose equal to the staff's proposed liquid annual limit (3 mrem/yr., total body) would result in an individual incurring an estimated increased risk of dying from cancer of 1 in 500,000 and an increased risk of a genetic effect to offspring over the next 5 generations of about 1 in 80,000. (The average risk to members of the U.S. population of dying from cancer is approximately 1 in 5 and the risk of genetic effects is about 1 in 17.)

4. If the cleanup is conducted in accordance with the staff's proposed effluent criteria, the total cumulative dose received by the entire population within a 50-mile radius of TMI-2 due to both gaseous and liquid releases would range from 10 to 30 person-rem for the entire cleanup. This is a small fraction (about .01%) of the background radiation dose received by the population from causes other than releases from TMI (population background radiation dose = 116 mrem/yr x 2.2 x 10^6 people = 255,000 person-rems). (See Sec. 10.3)

5. Although the number of truck shipments necessary to carry solid radioactive wastes to disposal sites will be large (ranging from about 350 to 1000), the shipments will be made over a long period and should cause little traffic congestion. Adherence to Federal packaging and shipping regulations will result in small radiation doses to those along the shipping route. If all TMI wastes are shipped to the furthest potential storage site, the estimated 700,000 persons who reside along the 2750-mile route might receive a cumulative population dose within the range of 20 to 50 person-rem. (See Sec. 10.3.3)

6. An individual onlooker who spends three minutes at an average distance of 3 ft from a truck loaded with radioactive waste might receive a dose of up to 1.3 mrem. This dose would increase the individual risk of dying from cancer by 1 in 1 million. The increased risk of genetic effects from the dose to offspring of the exposed individual is about 1 in 200,000. (See Sec. 9.5.1.2)

7. Radioactive fuel and high-specific-activity wastes from TMI-2 must be packaged and will have to be stored at the site temporarily until a suitable disposal site is established elsewhere. No significant environmental effects are expected from these activities.

8. From 1/2 to 1 acre of land at authorized disposal sites will be required for the low-level wastes from TMI-2. (See Sec. 9.5.1.3)

9. The expected consequences of credible accidents are small (below the requirement of 10 CFR Part 20 for normal operation). (See Sec. 10.4)

10. Resources that will necessarily be committed to the cleanup are materials of construction such as steel and cement, chemicals, organic resins, and other materials, none of which is in short supply in comparison to the annual U.S. production. (See Sec. 10.7)

11. Psychological distress caused by the accident and operations necessary to proceed with the cleanup has declined but there is a potential for temporary increases in distress as various cleanup activities are undertaken. (See Sec. 10.6)

12. Socioeconomic impacts include potential consumer avoidance of Chesapeake Bay seafood products that the public may believe are contaminated if processed water is released to the Susquehanna River, potential consumer avoidance of milk products sold directly to consumers following airborne releases of radioactivity, and the potential adverse market effect on residential property close to the transport route through Middletown, Pennsylvania, during the period of waste shipments from TMI. (See Sec. 10.6)

13. The differential monetary costs among suitable cleanup methods are small compared to the expected total costs of the entire cleanup and therefore do not constitute sufficient concern to affect a decision as to which alternatives should be chosen to accomplish the cleanup activities. The overriding considerations should be ensuring the public's health and safety and protection of the environment. (See Sec. 10.8)

12.2 Additional Conclusions

1. Existing methods are adequate, or can be suitably modified, to perform virtually all of the necessary operations without incurring significant environmental impacts. Where special tools or methods are found necessary for operations such as defueling, engineering expertise is available to cope with such requirements. (See Sec. 6.4)

2. An early decision to decommission TMI-2 would have very little effect on the choices of alternatives for the cleanup tasks because most of the same tasks must be performed in order to remove and dispose of the damaged fuel. (See Sec. 2.2)

3. If the damaged fuel and radioactive wastes are not removed, the Island would, in effect, become a permanent waste disposal site. The location, geology, and hydrology of Three Mile Island are among the factors that do not meet current criteria for a safe long-term waste disposal facility. Removing the damaged fuel and radioactive waste to storage sites that do meet all of the relevant criteria is the only reliable means for eliminating the risk of widespread uncontrolled contamination of the environment by the accident wastes. The staff has concluded that TMI should not become a permanent waste disposal site. (See Sec. 2.1)

4. Procedures have not yet been established for processing and disposal of high-specific-activity waste. Therefore, the staff regards the transfer of damaged fuel and high-specific-activity waste to facilities operated by the Department of Energy (where the national expertise exists) to be the most appropriate course of action for processing and final disposal of this material. (See Sec. 9.1.3.3)

5. The contaminated accident-generated water in the reactor building basement (sump) and in the reactor primary system cannot be left in its present condition and location if the cleanup effort is to proceed. Removal of this contaminated accident water will reduce the airborne and direct radiation levels in the building sufficiently to permit other cleanup operations to be accomplished with greater safety. (See Sec. 2.1.2)

6. Treatment of the contaminated accident water will transform the entrained radioactivity from its current mobile state to a more manageable form by concentrating and immobilizing the activity by an appropriate process. The cleanup activity will eliminate the risks associated with leaving the contaminated accident water radionuclide inventory in the mobile unprocessed state. (See Sec. 7.1)

7. A decision on the ultimate disposal of the processed water can be deferred until after the water has been processed. Then, the concentration of radionuclides remaining in the water will be low enough for the water to be stored safely onsite until the disposal decision is made. Processing the water to immobilize most of the radionuclides and storage of the processed water will not foreclose any reasonable options for disposition of the water or concentrated wastes. (See Sec 7.2.5.5)

8. The alternatives adopted for the various cleanup tasks will be those that keep the occupational doses as low as reasonably achievable. Delaying full cleanup will not appreciably lower the radiation fields (as a result of radioactive decay) or occupational doses. However, full and prompt cleanup would reduce the risks of uncontrolled radiation releases and would keep the doses to workers involved in cleanup tasks and to the public at a minimum. (See Sec. 10.2 & 10.3)

12.3 Benefits

1. The major benefit of the cleanup will be the elimination of the continuing risk of potential uncontrolled releases of radioactivity to the environment from damaged fuel or from the radioactive materials which are distributed throughout the primary system, the reactor containment building, and the auxiliary and fuel handling buildings. The radionuclides are also in the contaminated accident water in the reactor building basement and in the radioactive waste in temporary storage on the Island. These sources are a hazard because of the potential for uncontrolled radiation exposure to workers on the Island and to the local population. Removal of this hazard will relieve anxiety in some members of the local population and those dependent on the Susquehanna River and the Chesapeake Bay for a livelihood, for drinking water, or for recreation. The only way to eliminate this continuing hazard and anxiety is to clean up the facilities and remove the radioactive waste and damaged fuel to suitable storage sites. (See Chapters 2 & 4)

2. An incidential benefit that would accrue from the cleanup (and the ongoing studies that will be needed for planning and implementation) is additional knowledge that would be useful for reducing the risks and consequences of possible future accidents at nuclear power plants.

12.4 Cost-Benefit Balance

The staff concludes that on balance the above benefits and other considerations relative to the full decontamination, core removal, and disposal of the radioactive wastes from the March 28, 1979 accident at TMI-2 greatly outweigh the environmental costs of the cleanup activities. Until TMI-2 is largely decontaminated, there is a small possibility (which increases with time) of uncontrolled releases of radioactivity to the environment. Decontamination of the plant and disposal of the wastes will eliminate this possibility for potential harm to the public and workers at TMI, and will alleviate the attendant anxiety concerning radioactive releases from the plant. The staff therefore concludes that the full cleanup of the facility must proceed as expeditiously as is reasonably feasible, consistent with ensuring public health and safety and protecting the environment.

13. DISCUSSION OF COMMENTS ON THE DRAFT PROGRAMMATIC ENVIRONMENTAL IMPACT STATEMENT

Pursuant to 10 CFR Part 51, the Draft Programmatic Environmental Impact Statement (PEIS) related to the decontamination and disposal of radioactive wastes as a result of the March 28, 1979, accident at Three Mile Island Nuclear Station, Unit 2, was transmitted in August 1980 with a request for comments to the following federal, state and local government agencies:

Advisory Council on Historic Preservation
Council on Environmental Quality
Department of Agriculture
Department of Army, Corps of Engineers
Department of Commerce
Department of Energy
Department of Health and Human Services
Department of Housing and Urban Development
Department of the Interior
Department of Transportation
Environmental Protection Agency
Delaware Department of Natural Resources and Environmental Control
Maryland Department of Natural Resources
Maryland Department of State Planning
New Jersey Department of Energy
New Jersey Department of Environmental Protection
New Jersey Department of Public Utilities
Pennsylvania Department of Environmental Resources
Pennsylvania State Clearinghouse
Pennsylvania Public Utility Commission
Virginia Council on the Environment
Virginia State Corporations Commission
Susquehanna River Basin Commission
Dauphin County, Pennsylvania, Office of Emergency Preparedness
Londonderry Township, Pennsylvania, Board of Supervisors
Tri-County Regional Planning Commission
Commissioners of Dauphin County, Lancaster County, and York County Pennsylvania
Mayors of Middletown, Harrisburg, Goldsboro, Lancaster, York, Hershey, Lebanon, Hummelstown, Highspire, Camp Hill, Paxtang, Lewisberry, York Haven, Elizabethtown, Columbia, New Cumberland, Manchester, and Steelton, Pennsylvania
Commissioners of Baltimore and Hartford Counties, Maryland
Mayors of Baltimore and Havre de Grace, Maryland

In addition, a notice requesting comments from interested members of the public was published in the Federal Register on August 15, 1980, and about 2000 copies of the PEIS were subsequently mailed to individuals and organizations at their request. The comments received are reproduced in Appendix A of this Final PEIS, which is reserved solely for them.

The staff's consideration of the comments received and its disposition of the issues involved are reflected in part by revisions in the pertinent sections of this PEIS and in part by the following discussions. Where data corrections suggested in the comments have been adopted by the staff, these changes have usually been made without discussion here. The organization of this section corresponds generally to the ordering of the chapters of the statement; however, the discussions of comments on similar topics are grouped together. The comment letters to which these discussions apply are referenced by the numbers following the title of each response; these numbers are keyed to the letters in the Table of Contents in Appendix A.

13.1 GENERAL

13.1.1 Purpose and Scope of PEIS

13.1.1.1 The Purpose of Cleanup Operations (125)

The purpose of cleanup at TMI-2 is to remove from TMI-2 the potential for uncontrolled releases of radioactive material to the environment and the attendant risks to the health and safety of those members of the public residing in nearby communities. The cleanup would require the accomplishment of: building and equipment decontamination, reactor defueling and primary system decontamination, and processing, packaging, transportation and disposal of radioactive wastes.

13.1.1.2 Purpose of the PEIS (2, 33, 84, 86, 100, 107)

It is the purpose of the PEIS to present an overall study of the cleanup alternatives, including decontamination and disposal options for TMI-2, to assist the Commission in its decision-making on cleanup activities. The PEIS indicates viable choices for each of the cleanup steps. It is not, however, the purpose of the PEIS to predetermine which cleanup action should be chosen.

The scope and purpose of the PEIS were discussed with representatives of the President's Council on Environmental Quality prior to the drafting of the PEIS. It was agreed that in keeping with the purposes of NEPA to engage the public in the Commission's decision-making processes, and to focus on environmental issues and alternatives before commitments are made, no cleanup preferences should be predetermined.

The PEIS is written for the general public and government agencies with the purpose of involving the public in the NRC's decision making process regarding the cleanup of TMI-2 in accordance with the National Environmental Policy Act. Copies of this PEIS have been provided to government agencies and to all individuals who requested copies as well as to those who commented on the draft PEIS. In addition, copies of the draft PEIS were distributed to all who requested a copy during the numerous public meetings with interested individuals. Copies of this PEIS are available upon request. All comments and suggestions on the draft PEIS were evaluated by the NRC staff and all comments are addressed in this PEIS.

In order to encourage public involvement, efforts have been made to write this PEIS in laymen's terms such that the public reading the document can understand it. To this end, a glossary has been included at the beginning of this statement. In addition, a document, "Answers to Frequently Asked Questions About Cleanup Activities at Three Mile Island, Unit-2" (NUREG-0732) was prepared and made available to the public to explain the draft PEIS and to assist in preparing comments. The NRC staff also welcomes any questions from individuals referring to any item in the PEIS for explanation or clarification. Inquiries should be referred to the individuals indicated in the foreword of this statement.

13.1.1.3 PEIS Is Not a "Blueprint" for Cleanup (20, 121, 130)

It is not a requirement nor is it prudent that the PEIS should "serve as a blueprint" or be a "complete guide" to the cleanup activities. Many details of the decontamination process are not known because of limited access to the facility and its systems. The PEIS is intended to provide an overall evaluation of the environmental impacts that could result from the cleanup alternatives. As stated in the Commission's Statement of Policy and Notice of Intent to Prepare a PEIS (Appendix B), an overall study of the decontamination and disposal process will be "in keeping with the purpose of the National Environmental Policy Act to engage the public in the Commissions' decision-making processes, and to focus on environmental issues and alternatives before commitments to specific cleanup choices are made".

13.1.1.4 Supplements to the PEIS (20, 59, 85, 100, 115)

To avoid as much as possible any segmentation of the evaluation of environmental impacts, the PEIS presents an overall description of all of the main cleanup activities along with a discussion of alternatives and the environmental impacts of these alternatives. However, there are many uncertainties regarding the cleanup operations. For example, the precise condition of the reactor core will not be known until the reactor vessel has been opened and the core inspected. If, when more information becomes available, the effects of any proposed activities are found to be significantly beyond the scope of the assessments made in this statement, appropriate supplements will be issued. These supplements will not be isolated from the PEIS, but instead, they will take into account any additional effects on alternatives previously discussed in the PEIS (e.g., change of schedule) and the cumulative environmental impact of the cleanup operations (e.g., total occupational dose, total radioactivity releases). Furthermore, the public will be informed of any supplements and they will be made available for public comment.

13.1.1.5 Ultimate Disposition of Unit-2 (20, 31, 60, 100, 124)

Whether or not Unit 2 will be decommissioned or restored to a condition acceptable for licensed operation is not within the scope of the PEIS. Discussions concerning the ultimate disposition of Unit 2 are presently premature since the licensee has not made any proposals to reactivate or decommission the reactor. In either case, the facility will have to be cleaned-up. However, to determine if there would be any effects on the way the cleanup could be accomplished, the staff did evaluate the effects on an early decision to decommission TMI-2. This is discussed in Section 2.3 and detailed in Appendix U of this PEIS.

13.1.1.6 NRC's Role in Prevention of the TMI-2 Accident (67)

Discussion of NRC's role in preventing the accident at TMI-2 or at any operating reactors is beyond the scope of this Statement. This topic is discussed in several of the various reports of groups which investigated the accident, in particular those of the President's Commission (Kemeny) and the NRC's Special Inquiry directed by Mitchell Rogovin.

13.1.1.7 Restart of Unit-1 (27, 31, 102, 124)

Issues concerning the restart of TMI-1 are addressed in a separate environmental review. The staff considers the restart of TMI-1, if authorized, to be wholly independent of the TMI-2 decontamination process.

13.1.1.8 Cleanup Alternatives Involving Other Agencies Included (124)

Cleanup alternatives that may involve another agency other than the NRC (e.g., EPA, DOE) are included in the discussion presented in this Statement.

13.1.2 Content and Organization

13.1.2.1 Format of the Final PEIS (30, 33, 85, 86, 100, 110, 32, 115, 116, 123)

While the draft statement had the benefit of enabling the reader to follow the chronological sequence of the cleanup activities, it also had the disadvantage of scattering significant information on a particular issue, such as water processing, throughout various parts of the draft. As a result, Sections 5 through 8 are reorganized to group like activities together.

Further efforts have been made to write the PEIS in laymen's terms so that the document can be easily understood. Toward this end, the glossary has been extended.

13.1.2.2 Inadequacy of Summaries (52)

While some additional technical information has been added to appropriate parts of the summary, in general, the summary has been simplified so that it is more readily understandable by the public.

13.1.2.3 Numbers and Calculations in Draft PEIS are not "traceable" (32, 35, 37, 79)

To the extent practicable, sources of numerical calculations have been identified and cited throughout this Statement.

References which point to such items as international reports, reviews, etc. are generally available at specialist libraries maintained at nuclear energy facilities or at major universities. Page numbers have also been added to many of the reference citations so that specific points of interest can be located easily.

"Handouts to presentation of GPU personnel at staff site visit, January 1980" is available in the NRC public document rooms. An update of the cost and schedule information, "TMI-2 Recovery Program Estimate, August 1, 1980" has been transmitted to the NRC by the licensee and is also available for public inspection at the NRC public document rooms.

13.1.2.4 Organization of Page Numbers (100)

The page numbers are organized by sections and are preceded by a section number. Individual sections are grouped according to major topics, making it easier to locate the topic of interest.

13.1.2.5 Basis for Sludge Volumes (32)

The volume of sludge varies somewhat at each location when fluids are circulated. The movement of fluid transports sludge to the filters where it is trapped. In areas such as the Auxiliary and Fuel Handling Building (AFHB) Sump, sludge levels may increase due to decontamination efforts that flush solids to the sump, and they may decrease as water and some sludge are pumped out of the sump.

The AFHB Sump was tested by licensee personnel with long rods to determine the depth of the water above the sludge. Using that water depth and the as-constructed dimensions of the Sump, the sludge volume was calculated to be 200 ft^3. The measurement and calculations gave the sludge volume at one point in time and it could increase or decrease somewhat due to operations. For this reason sludge volume is usually an estimated value.

In those tanks that have not been tested by personnel, the sludge volumes were estimated conservatively using the standpipe height and the tank dimensions. This was done for the Miscellaneous Waste Holdup Tank and the Sump Tank. Experience has shown that tanks involved in numerous fluid transfers have little or no sludge buildup. The estimates for the Reactor Coolant Bleed Tanks and others used in the management of liquid wastes were based on this experience. The larger sources of sludge, such as the AFHB Sump, have a better defined sludge volume than most of the tanks. Therefore, the overall uncertainty in total sludge volumes given is not large, probably at most 5%.

13.1.3 Licensee's Plans and NRC Actions

13.1.3.1 Licensee's Obligation to Cleanup TMI-2 (100)

The Commission, in its Statement of Policy of September 26, 1980, has clearly stated the need for the licensee to continue with its cleanup activities and maintenance of the facility to protect the health and safety of the public. Although the NRC does not have the authority to determine how the licensee should allocate its resources, the licensee has the obligation to complete the cleanup of TMI-2.

13.1.3.2 NRC's Responsibilities Regarding the Cleanup Activities (12, 38, 66, 67, 101)

By the Atomic Energy Act of 1954 as amended, the NRC is mandated to protect the health and safety of the public from activities under its license. The NRC is the independent agency established by the Congress, staffed with knowledgeable experts to monitor and oversee the licensee's cleanup activities to ensure the safety of the public. All cleanup operations at TMI-2 have to receive the authorization of the NRC. In addition, through its TMI Program Office onsite, the NRC oversees all licensee cleanup activities to ensure that operations are implemented according to NRC-approved plans and in compliance with NRC-approved limits, procedures and Orders.

13.1.3.3 Authorization of Cleanup Activities (30, 67, 70, 125)

When the NRC receives specific cleanup proposals from the licensee, the staff will review them vis-a-vis the environmental evaluations in the PEIS and with respect to safety requirements. After these reviews, the NRC will either proceed to act upon the proposal or conduct additional environmental and safety reviews, if necessary, before making a decision on approval.

13.1.3.4 PEIS Not to Predetermine Cleanup Actions (64, 75, 80, 90, 100, 107, 124)

Viable alternatives for cleanup are presented and evaluated in this statement. However, it is not the purpose of this Statement to predetermine which of the viable alternatives should be accepted. Only when the license submits an actual cleanup proposal will the NRC staff review the proposal and act accordingly or make recommendations to the Commission, if appropriate. Based on the staff evaluation and recommendation, the comments from the public and other agencies on the PEIS, the Commission will make decisions on the authorization and acceptability of the proposed activity.

13.1.3.5 Need to Prepare PEIS (116)

The NRC recognizes that the cleanup must proceed expeditiously. However, it is also important that potential environmental consequences of cleanup alternatives as well as public comments and suggestions be considered prior to cleanup decisions being made.

13.1.3.6 Cleanup Decisions prior to the PEIS (16, 20)

NRC decisions on previous cleanup operations were made following extensive reviews, including evaluation of feasible alternatives. For each of the major cleanup operations (e.g. decontamination of the reactor-building atmosphere and processing of contaminated wastewater by the EPICOR-II system), the NRC staff issued separate environmental assessments which were circulated for public comment. In each case, all feasible alternatives were evaluated and comments and suggestions from the public and interested groups were considered before the decisions were made. These actions were consistent with the requirements of the National Environmental Policy Act; however, it was not prudent to wait for the completion of the PEIS to carry out these cleanup operations because of safety and environmental considerations.

The construction of the Submerged Demineralizer System to clean up the contaminated water in the reactor building is proceeding at the risk of the licensee, without NRC authorization of its usage.

13.1.3.7 Safety Evaluation of Cleanup Proposals (59, 91)

When the licensee submits a proposed cleanup action, the staff will then review it and perform a safety evaluation in addition to evaluating environmental considerations. For example, safety precautions and backup systems will be reviewed for adequacy to ensure that the proposed action will be safe to implement. Appropriate technical specifications will be required for the implementation of the approved activities, along with approved procedure for implementation. The NRC staff also will maintain an inspection program to ensure that the approved procedures are followed during cleanup operations.

13.1.3.8 Decision Procedure Prior to Authorization of Processed Water Disposition (32, 39, 40, 42, 62, 77, 125, 130)

Pending completion of the final PEIS, the licensee has not made any proposal on the action to be implemented for the dispostion of processed water. In fact the licensee indicated that no proposal will be made before 1982. When such a proposal is received by the NRC for approval, the NRC staff will evaluate the proposed action against the alternatives already assessed in the PEIS and make recommendations to the Commission. The decision of the Commission will be based on the environmental assessment of the alternatives contained in the PEIS, the staff's recommendation, the comments and suggestions from the public and from other government officials, and the Commission's own assessment. Potential environmental impacts, such as the quality of public water and potential impacts on fish and seafood in the Susquehanna River and the Cheasapeake Bay, are certainly major considerations to be included in any decisions.

13.1.3.9 The Pace of Cleanup Activities (50, 70, 124)

While the need to expeditiously proceed with the cleanup is desirable, it is also important that the cleanup proceed in such a manner as to minimize any risks to the health of workers and nearby residents i.e., only after careful assessments of the potential environmental impacts from the cleanup activities have been made.

13.1.3.10 Cleanup Delays (67)

It is unlikely that short delays would appreciably affect the environmental impact assessment. If long delays that may significantly impact on the environment occur because of the financial condition of the licensee, the NRC will act to ensure that the health and safety of the public is ensured and that the cleanup proceeds without further adverse impact. If the delay is because of new conditions which have an environmental impact significantly beyond the scope discussed in this PEIS, supplement(s) to the PEIS will be prepared and issued for public review and comment.

13.1.3.11 Cleanup Activities Planned for 1981 (114, 124)

Major progress on the Unit-2 cleanup is planned for 1981. Some examples of cleanup activities the licensee plans for 1981 include: (1) The establishment of long-term decay-heat removal for maintaining safe reactor shut down; (2) continue cleanup of auxiliary building individual areas and equipment; (3) continue reactor building entries for data acquisition; (4) cleanup of reactor building sump water; (5) engineering and construction of support facilities required for cleanup and waste dispostion.

13.1.3.12 NRC Policy Requires Cleanup to Proceed (51, 52, 59)

On September 29, 1980, the NRC issued a Statement of Policy with regard to the requirement of the licensee to proceed with the cleanup. It states that "The Commission will not excuse Met Ed from compliance with any order, regulation or other requirement imposed by this Commission for purposes of protecting public health and safety or the environment." Should the licensee fail to meet its obligation for financial reasons, the NRC has, under existing laws, the authority to step in and to act to ensure that the cleanup proceeds in a timely manner.

13.1.3.13 Uncertainties on Future Cleanup Proposals (100)

To the extent known, the cleanup conditions have been evaluated. Where conditions are not yet known, the environmental review was performed on the basis of the best information available with due acknowledgment of uncertainties. In some instances this necessitated reliance on "best-case" and "worst-case" evaluations. Upon submission of specific proposals by the licensee, any necessary further environmental review undertaken by the staff will include all information then available.

13.1.3.14 Systems Shared with Unit 1 (51, 101)

There is no Unit 1 system presently planned to be shared with Unit 2 that may affect the safety of Unit-1 operation or Unit-2 cleanup. This separation includes the facilities to store waste water.

13.1.3.15 Quality Assurance Regarding Cleanup Personnel (66)

The licensee's quality assurance program provides that all cleanup workers should be properly trained for the particular cleaning tasks assigned. The radiation protection plan also requires workers entering the radiation area to have a Radiation Work Permit (RWP). Prior to receiving the RWP, the worker must be trained in the basics of radiation hazards and protection techniques. The NRC also reviews all cleanup plans, equipment, methods and procedures prior to their implementation to ensure personnel safety and adequate protection of the environment.

13.1.3.16 Cost of Cleanup Not a Major Factor for NRC Decisions (32, 64, 66, 67)

The licensee's proposed actions will be independently evaluated by the NRC staff according to the mandated function of the NRC to protect the health and safety of the public. Estimated differential costs between the alternatives are provided in the PEIS. However, these cost estimates will not be a major factor in the evaluation of the licensee's proposed cleanup activities.

13.1.4 Public Concerns and Participation

13.1.4.1 Public Participation prior to the Draft PEIS (11)

The NRC staff, as directed by a Commission Order dated May 25, 1979, prepared an "Environmental Assessment--Use of EPICOR-II at Three Mile Island, Unit 2" (NUREG-0591) dated August 14, 1979. This document was circulated to appropriate government agencies and the general public for a 30-day comment period. The staff provided a discussion of the public comments on the EPICOR-II environmental assessment, as well as its recommendation to the Commission on October 4, 1979. The Commission, after considering NUREG-0591, public comments and the staff's recommendation, authorized operation of EPICOR-II on October 16, 1980.

The public also took part in the decision-making process which led to the decontamination of the TMI 2 reactor building atmosphere. Approximately 800 comments were received from members of the public, state and federal agencies in response to the circulation of the draft environmental assessment (NUREG-0662) on the subject in March, 1980. At the request of the Governor of Pennsylvania and members of the public, the NRC extended the public comment period on draft NUREG-0662 for 30 days beyond the original deadline of April 17, 1980. Nearly 50 meetings were held with local officials, organizations and members of the public to discuss the issue. The NRC staff considered and responded to public comments on the draft in its final environmental assessment dated May 1980, and the Commission was briefed in two public meetings. The Commission's decision was given in its June 12, 1980 order to make a controlled purge of the reactor building to the atmosphere.

Public scoping meetings on the proposed content of this PEIS were held in the Harrisburg, Middletown and Baltimore areas during January, February and March of 1980 prior to completion of the draft PEIS. These scoping meetings are discussed in Section 1.4.

13.1.4.2 Opportunities for Public to Comment on PEIS (11, 16, 20, 124)

Members of the public have had the opportunity to participate in commenting on the PEIS in several ways:

An extended, 90-day comment period, which ended on November 20, 1980, was provided for interested individuals and groups to submit their comments on the draft PEIS in writing. The NRC staff also met with numerous groups in Pennsylvania, in the area around Three Mile Island, and in Maryland to solicit comments on the draft PEIS. Individuals were notified of these meetings, which were publicized and were open to public participation. Any portion of the PEIS could also have been discussed with staff members at the NRC-TMI Middletown office.

13.1.4.3 Public Hearing on PEIS (16, 20, 60, 64, 100, 121)

The NRC staff does not believe that a formal, adjudicatory hearing is required in connection with the PEIS, nor has the Commission itself deemed it otherwise appropriate at this time. When the licensee submits a specific proposal regarding a particular cleanup activity to the NRC which requires the issuance of a license amendment, appropriate notice will be given to the public. At that time, a hearing may be requested in accordance with Section 189 of the Atomic Energy Act and the Commission's implementing regulations. This process is in accordance with the NRC's statutory obligations and is consistent with Section 1506.6(c) of The President's Council on Environmental Quality (CEO) regulations.

13.1.4.4 Recirculation of PEIS Because of Cost Data (64, 79, 85, 94, 100, 104, 130)

A number of commenters requested that the draft PEIS be supplemented and recirculated in light of the absence of cost data and the development of newer or additional information for inclusion in the final PEIS. The guidelines of the Council on Environmental Quality (CEQ) state that a supplement must be prepared by an agency if: (i) the agency makes substantial changes in the proposed action that are relevant to environmental concerns; or (ii) there are significant new circumstances, or information relevant to environmental concerns and bearing on the proposed action or its impacts. (40 CFR § 1502.9(1)). Cost data does not, in the context of the PEIS, constitute an "environmental concern" so as to give rise to the requirement to supplement and recirculate the draft PEIS. Cost data is relevant primarily for purposes of comparing the overall desirability of a chosen course of action with reasonable alternatives thereto. Economic considerations are not of foremost importance in terms of protecting the public health and safety. The CEQ guidelines further provide that the weighing of the merits and drawbacks of the various alternatives need not be displayed in a monetary cost-benefit analysis and should not be when there are important qualitative considerations. (40 CFR § 1502.23.) Similarly, no "new" information in the final PEIS is "significant" in terms of its impact on the evaluation of the proposed cleanup. Rather, it consists of revisions in the textual material as the result of comments received on the draft PEIS, minor corrections and reflects the acquisition of more timely information. The National Environmental Policy Act expressly contemplates that changes in a draft environmental statement will result from the comment process. That is its purpose. This fact, coupled with the overriding importance of proceeding with the cleanup, mitigates against the provision of an additional circulation process. The Staff believes that the draft PEIS has afforded the public with a fair opportunity to comment on aspects of the cleanup process to the extent presently known.

13.1.4.5 Influence of Comments on Decisions (66, 69)

The comments from the public form a substantial factor in the NRC's decision-making process. Each comment receives careful review by the NRC staff and all pertinent comments are considered in its preparation of this Statement.

13.1.4.6 Independent Scientific Review (16, 30, 66, 67, 99, 101)

The NRC is staffed with scientists, engineers and experts to evaluate all phases of cleanup operations. Other government agencies, e.g., the EPA and the DOE, also participate in the review of specialized areas of the cleanup, e.g., monitoring the environment and disposal of nuclear fuel materials and high-activity radioactive wastes. In addition, the Commission has established and funded an Advisory Panel to consult with, and provide advice to the Nuclear Regulatory Commission on major activities required to accomplish the safe cleanup of the TMI-2 facility. The NRC Advisory Committee on Reactor Safeguards (ACRS) is available to provide the Panel technical assistance, as necessary, related to the cleanup. As originally established, the Advisory Panel consists of twelve members: three members of the public residing in the vicinity of TMI; three members from the independent scientific community; three members of the Pennsylvania state officials and three members from local government officials.

13.1.4.7 Future Public Participation (20, 39, 40, 72, 102, 114)

If future proposed actions or data related to conditions in the reactor are within the scope of potential environmental impact discussed in the PEIS, then public participation in the assessment would have already taken place by means of the public meetings and comments received on the PEIS. But, if future information, or the proposed cleanup actions, are significantly different from those alternatives addressed in the PEIS, appropriate supplements to the PEIS will be prepared and the public will be provided opportunities to participate in these new environmental assessments. Public meetings may be scheduled. For licensee actions that would require an amendment to the license, an opportunity to request and participate in a formal adjudicatory hearing will be provided, consistent with the Atomic Energy Act of 1954 as amended, and the NRC's implementing regulations.

13.1.4.8 Comments on the Final PEIS (41, 92)

Public comments on the final PEIS would be considered by the NRC staff if the comments are on new information in the final PEIS not previously presented in the draft PEIS and (2) such new information is significant in terms of environmental impact.

13.1.4.9 Future Information on Cleanup (88)

Relevant information on the cleanup will be provided to the public throughout the cleanup process in several ways; 1)status reports will continue to be issued by the NRC; 2) safety evaluations by the NRC staff on specific licensee cleanups proposals will be available to the public and 3) decisions by the Commission will also be published documents available to the public.

13.1.4.10 Local Referendum on Waste Disposal Alternatives (67)

A local referendum on waste disposal alternatives has not been planned.

13.1.5 Decontamination Experience at Other Nuclear Facilities

13.1.5.1 Experience with Decommissioning Reactors (69, 100)

A number of nuclear reactors have been decontaminated and decommissioned in the U. S., although none were identical to TMI-2. Of these, perhaps the most notable is the Experimental Breeder Reactor-I (EBR-I) which operated at the Idaho National Engineering Laboratory from 1951 to 1964. EBR-I was decommissioned and turned over to the U. S. Department of Interior. (It was subsequently declared a National Historical Landmark and has been open to the public for tours since 1975.) Other examples of decommissioned reactors include reactors at Elk River, Montana, several at the Hanford Reservation in the State of Washington, and the Enrico Fermi I Plant near Detroit, Michigan. The power reactor at Shippingport, Pennsylvania is currently undergoing decommissioning.

Significant experience exists in surface decontamination and cleanup of contaminated water. Most proven decontamination methods are easily applied at TMI-2. Even though there is more surface area and contaminated water volume at TMI-2 than has been experienced at other facilities, the technology and methods are basically the same.

Because of the uncertainties with respect to the condition of the damaged core, of the major cleanup steps, only fuel removal may require techniques that are relatively new, although this is doubtful. Prior experiences with damaged core removal with the Canadian NRX and NRU reactors (Section 1.5.3) and the Enrico Fermi I reactor indicate that the operation can be successfully completed without major delays or the need to develop radical new technologies.

The fact that large scale decontamination is not factored into designs of commercial nuclear power plants does not mean that the decontamination of TMI-2 would be an experiment requiring untested technology. For example, the amount of krypton-85 at TMI-2 is a major difference between TMI and other decontamination situations. The krypton problem was solved by purging with no significant environmental impact. The amount of water and its contamination level in the containment sump at TMI-2 is, relatively high. However, decontamination can be accomplished without significant departure from existing technology. The decontamination will result, however, in a larger quantity of radioactive waste. Methods for handling and disposing of this waste are also within existing technology.

13.1.5.2 Information on Dresden-I Decontamination (81, 100)

The NRC staff has reviewed the information in the Environmental Impact Statement prepared for the Dresden decontamination. Information germane to the TMI-2 decontamination has already been considered in this Statement.

13.1.5.3 Decontamination Experience at Chalk River (100, 115)

In each of the two incidents of overexposure at Chalk River, the over-exposures (in excess of 5 rem) were incurred by individuals who were not normally radiation workers.[1,2] At TMI-2, all cleanup workers entering radiation areas are required to receive basic training in radiation protection. In addition, the health physics program and the radiation protection plan procedure at TMI-2 would provide added assurance for preventing incident of over-exposure for cleanup workers. With respect to the contaminated water in the NRX incident in 1952, the water was pumped from the reactor basement for about 1-1/4 miles to trenches which had been cut into the sandy soil. These trenches were about 20 feet wide, 10 feet deep, and about 1000 feet long.[3] Follow-up studies concerning the disposition of the radioactiity have been conducted by personnel of the Environmental Research Branch of the Chalk River installation.[3-6] Initially, most of the radionuclides were trapped near the surface of the sand.[3,4] After several years, some radioactive material migrated underground from the disposal area toward a swamp area. These radionuclides had been transported down to the water table by leaching and percolation and subsequently migrated with the ground water.[6] Further investigations by the Environmental Research Branch have

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[1] A. J. Cipraini, "Health and Safety Activities in Reactor Operations and Chemical Processing Plants," International Conference on Peaceful Uses of Atomic Energy, Vol. 13, 263-265, UN Publications, Geneva 1956.

[2] E. O. Hughs and J. W. Greenwood, "Contamination and Cleanup of NRU," Nucleonics, Vol. 18, No. 1, 76-80, January 1960.

[3] C. A. Mawson, "Waste Disposal in the Ground," International Conference on Peaceful Uses of Atomic Energy, Vol. 9, 676-678, 696, UN Publications, Geneva 1956.

[4] G. C. Butler et al., "Health and Safety in Canadian Operations," 2nd UN Conference on Peaceful Uses of Atomic Energy, Vol. 21, 19-24, UN Publications, Geneva 1958.

[5] P. J. Parsons, "Movement of Radioactive Waste Through Soil," AECL-1038, June 1960.

[6] P. J. Parsons, "Movement of Radioactive Waste Through Soil," Second Ground Disposal of Radioactive Wastes Conference, TID-7628, March 1962.

attempted to characterize the migration of activity.[5,6] Strontium-90 was the most prominent radionuclide identified and its transport in soil was such that it would require about 150 years before releases to the environment, at which time the amount will have sufficiently decayed to present a minimal risk to the public.[6] Review of the experience gained and estimates of the population exposure due to the waste disposal operations at Chalk River have been published.[7,8] With respect to the TMI-2 cleanup, the treatment of the contaminated water will differ significantly from that at Chalk River. Regulations applicable to TMI-2 do not allow contaminated water to be released. The water will be extensively decontaminated prior to disposal from the site.

13.1.5.4 Limited Cleanup Experience (100)

The apparently contradictory nature of some statements in the draft PEIS can be easily resolved. The staff concludes in the PEIS that "the basic technologies for decontamination are well established and that available techniques can be modified to suit the conditions at TMI-2." The key in the above statement is the term "basic technologies" which refers to the general level of scientific knowledge about decontamination, and the industry's ability to apply that knowledge to the practical purpose of decontaminating TMI-2.

In Section 1.5.2, a statement is made regarding limited experience with high level decontamination of large areas of interior building surfaces. This is not contradictory to the statement above because the basic technology to accomplish high level decontamination of large areas is well established, but has not been extensively required before TMI-2. Limited experience also does not mean no experience. Three examples are given in Section 1.5.2, with the Canadian NRX Reactor a major one. While TMI-2 is a larger decontamination effort, "available techniques can be modified to suit conditions at TMI-2."

The statement on the limited experience with removal of damaged fuel and core components, and the need for specific techniques for TMI-2 is also not contradictory. The basic technology is well established and we are confident that the TMI-2 core can be cleaned up by using that basic technology to develop TMI-2 specific techniques, methods, and tools. Each decontamination effort is in a sense one-of-a-kind, having its own specific problems that must be overcome. It is well established basic technology that allows industry to accomplish these cleanup activities.

The statement regarding chemical decontamination experience to remove fuel failure debris (Sec. 1.5.4), is also not really contradictory. Again, very limited does not mean non-existent, and examples are given where chemical means have been employed. In fact, Section 1.5.4 concludes in the third paragraph that chemical "fuel removal technology is available for application at nuclear plants." This again refers to the existence of the basic technology needed to accomplish the decontamination.

13.1.6 Regulatory Requirements and Authorization

13.1.6.1 Current TMI-2 Licensing Conditions (20, 38, 79, 116)

Since the accident, the Unit-2 license has been amended by order of the Office of Nuclear Reactor Regulations (NRR). On July 20, 1979, NRR issued an order for Modification of License which suspended the licensee's authority to operate the facility except in its shutdown condition and

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[5] P. J. Parsons, "Movement of Radioactive Waste Through Soil," AECL-1038, June 1960.

[6] P. J. Parsons, "Movement of Radioactive Waste Through Soil," Second Ground Disposal of Radioactive Wastes Conference, TID-7628, March 1962.

[7] I. L. Ophel, "Environmental Consequences of Radioactive Waste Disposal," Pollution and Our Environment, Vol. 1, Background Paper A4-4-2, Canadian Council of Resource Ministers, 1966.

[8] C. A. Mawson and A. E. Russel, "Canadian Experience with a National Waste Management Facility," Management of Low- and Intermediate-Level Radioactive Wastes, IAEA, Vienna, 1970.

require that pending further amendment of the license (DPR-73), the licensee shall maintain the facility in a shutdown condition in accordance with the approved operating and contingency procedures for the facility. On February 11, 1980, by Order of the Director, NRR, a new set of formal license requirements was imposed to reflect the post-accident condition of the facility. These requirements ensure the continued maintenance of the current safe, stable, long-term cooling condition of the facility. These requirements were set forth in a new set of proposed Technical Specifications contained in an attachment to the Order.

Future license amendments on specific proposed cleanup activities will be based on the environmental assessments contained in the PEIS or any supplements thereto, together with an appropriate safety evaluation of the proposed activity.

13.1.6.2 Technical Specification Unrelated to Cleanup (100)

The Technical Specifications issued on February 11, 1980 which are subject to the pending litigation address maintenance of the reactor in a safe shutdown condition. These Technical Specifications are unrelated to the decontamination process and are not included in this statement. However, please note that appropriate regulatory criteria and proposed Technical Specifications pertaining to the decontamination process are included in Section 1.6 and Appendix K.

13.1.6.3 As Low As Reasonably Achievable (ALARA) Principle and Cleanup Procedures (52, 125)

In carrying out the cleanup operations, the licensee is required to comply with regulations set forth in 10 CFR Parts 20 and 50 in maintaining radioactivity releases to the environment and worker exposure as low as reasonably achievable (ALARA). Specific numerical limits to implement the ALARA principle will be developed for the cleanup programs. A more detailed discussion of the proposed criteria for radiological effluents is in Section 1.6.3.2. When cleanup activities are being authorized, the technical specifications and operating procedures will be reviewed to assure that the ALARA principle will be met during implementation of cleanup operations.

13.1.6.4 Cleanup Criteria (13, 50, 100, 124)

Discussions covering the criteria currently in effect at TMI-2 as well as proposals for future criteria are presented in Section 1.6. The interim criteria established for radiological effluents are based on data-gathering and maintenance operations. These criteria were approved by the NRC on April 7, 1980, and are presented in Section 1.6.1.6. Some criteria affecting the TMI-2 are expected to be developed over a period of time as a result of the issuance of this Statement. These are discussed in Section 1.6.3. These criteria consist of: (1) future radioactive waste disposal standards (Section 1.6.3.1); (2) proposed criteria for radiological effluents from decontamination activities (Section 1.6.3.2); and (3) acceptable removable surface contamination levels (Section 1.6.3.3) for unrestricted access.

13.1.6.5 Amendments to Technical Specifications (100)

Proposed additions to Technical Specifications for the TMI-2 cleanup program are included in Appendix R. Any additions or modifications to the existing TMI-2 Licensing conditions (Technical Specifications) implementing TMI-2 cleanup operation will be added to the TMI-2 operating license by means of the license-amendment process. This process includes NRC review and evaluation for safety and environmental concerns, notification of the amendment and provides an opportunity for interested persons to request a hearing.

13.1.6.6 Redundancy of Technical Specifications on Dose Estimates and Reporting (50)

Proposed Technical Specification R.1.3 has been revised to eliminate the redundancy in dose estimate and reporting requirements.

13.1.6.7 The Need to Perform Dose Estimates During Cleanup (50)

The bases for the proposed technical specifications are described in Appendix R. Dose calculations will provide NRC with necessary information as the cleanup progresses to assess the environmental impact due to cleanup activities, to maintain radiation doses to as low as reasonably achievable. As more experience is gained with each program, calculations for later programs can more accurately be made than the dose projected in this Statement.

Dose estimates accompanying proposals of cleanup activities are necessary for the staff to ascertain the potential environmental impact, both during normal operation and under unplanned conditions. Dose estimates are also required for the review of on-going and future cleanup programs. The requirements for dose estimates are not different from current requirements for operating reactors to assure compliance with the As low As Reasonably Achievable (ALARA) principle of Appendix I to 10 CFR 50. The staff is aware that the estimated doses may be very low for some of the alternatives. The estimates for a particular cleanup proposal would still be necessary to form the basis of comparison with other alternatives and to ensure that the ALARA principle is met.

13.1.6.8 The Application of Appendix I Criteria (100)

Appendix I is not developed on cost-benefit consideration based on electricity generation. Rather the numerical values are based on the requirement of radioactive effluent treatment systems meeting the objective for maintaining radiation doses to the public at levels as low as reasonably achievable (ALARA). By the same analogy, the benefit gained from cleanup is not measured by generation of electricity but rather by the need to ensure protection of public health and safety and the environment. The same ALARA principle should therefore be applicable to cleanup operations.

13.1.6.9 Potential Conflict Between NRC Requirements and Other Constraints (80)

A list of constraints (settlements with local government agencies, discharge permits of the Commonwealth of Pennsylvania, etc.) are presented in Section 1.6.2. With respect to the potential of future conflicts with requirements of other agencies, a Statement of Policy issued by the Commission on September 29, 1980, illustrates the general principle under which the NRC written in jurisdiction . . . creates an irreconcilable conflict with NRC requirements which have been imposed on Met Ed or which may be imposed in the future. We wish to state clearly, however, that in the event of any such conflict, NRC health, safety and environmental requirements must supersede state agency requirements that result in a lesser degree of protection to the public."

13.1.6.10 Emergency Preparedness Plan (20, 32, 51, 46, 99, 115)

The scope of the PEIS does not include discussion on emergency planning. However, the following information is available. Nuclear powerplant licenses are required to submit their emergency preparedness plans for NRC staff review and evaluation against criteria of NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants." On June 10, 1980, the licensee submitted to the NRC for review the emergency preparedness plan for TMI Unit 1 ("GPU Corporation Emergency Plan for TMI Nuclear Station, Unit 1"). A Safety Evaluation Report (SER) to address the licensee's proposed plan for Unit 1 will be issued by the NRC staff prior to the conclusion of public hearings currently being held on the proposal to restart Unit 1. The Emergency Preparedness Plan for Unit 2 is being prepared by the licensee and is expected to be submitted to the NRC in the near future. After receiving the proposed plan for Unit 2, the NRC staff will review and evaluate the proposal and discuss its findings in SER.

13.1.6.11 Violations of the Clean Water Act and NEPA (20, 52, 67, 91, 114)

The Federal Water Pollution Control Act authorizes discharge of pollutants only in conformance with its provisions and those of the National Pollutant Discharge Elimination System (NPDES) Permit issued by the U. S. Environmental Protection Agency or its delegate (i.e., the Commonwealth of Pennsylvania). NPDES Permit No. PA0009920 (dated July 12, 1977) was issued to Metropolitan Edison Company by the USEPA for Three Mile Island Nuclear Station. Special conditions numbers 7 and 8 in Part III of that NPDES Permit require radiological effluents to meet NRC requirements as follows:

"All limitations and monitoring requirements for liquid radioactive waste discharges shall be in accordance with the Nuclear Regulatory Commission regulations as set forth in 10 CFR Part 20 and 10 CFR Part 50" [Special Condition No. 7]. "The conditions of this permit shall in no way supersede the mandatory requirements for operation of Nuclear Power Plants imposed by the Nuclear Regulatory Commission" [Special Condition No. 8].

If the licensee proposes to release the processed water into the Susquehanna River and if the NRC authorizes such a proposal, neither the intent nor any of the provisions of the Act would be violated, since the NPDES Permit would be followed.

The National Environmental Policy Act of 1969 (NEPA) (PL 91-190), established national policy for considering environmental consequences in decision making. NEPA also established procedural requirements for the consideration of environmental matters in decision making. A major procedural requirement is the preparation of an environmental statement for Federal actions which have potentially significant environmental impacts, with a consideration of alternatives to a proposed action. This Statement was developed in conformance with NEPA, as stated on page 1-1: (the PEIS is) ". . .in keeping with the purposes of the National Environmental Policy Act to engage the public in the Commission's decision-making processes, and to focus on environmental issues and alternatives before commitments to specific cleanup choices are made."

NEPA does not specifically prohibit discharging nor does it establish effluent limitations for radioactive wastes. Section 104 of NEPA states that "Nothing in Section 102 or 103 [procedural sections] shall in any way affect the specific statutory obligations of any Federal agency (1) to comply with criteria or standards of environmental quality. . .or (3) to act, or refrain from acting contingent upon the recommendations or certification of any Federal or State agency." With respect to this Statement, NRC is: involving the public in its decision-making process; considering alternative courses of action and the environmental consequences of each; and, is requiring any radioactive discharges under consideration or examination to meet all effluent standards for protection of the public health and safety in accordance with the FWPCA and NPDES permit.

13.2 MAJOR PROGRAMMATIC ALTERNATIVES

13.2.1 General Alternatives

13.2.1.1 "Entombment" of Fuel and Radioactive Wastes in the Reactor Building (4, 104, 107, 114, 116, 121, 124, 125)

As pointed out in Sections 2.1 and 2.2, entombment of fuel and radioactive wastes in the reactor building would require changes in current national policies and federal regulations, for it would convert the TMI-2 site to a high-level waste repository. TMI-2 does not qualify as a repository under current policies and regulations.

The advantages of such entombment must be weighed against the disadvantage of creating a long-term hazard and source of public stress and anxiety that would continue for several hundred years. The staff regards this disadvantage to be so serious that it more than offsets any gain from sealing up the reactor building with the fuel and contamination inside until the radioactivity has decayed to "safe" levels. Decommissioning alternatives for TMI-2 are evaluated in Section 2.2 and discussed in detail in Appendix U.

13.2.1.2 A Wider Look at Clean-up Alternatives? (64, 100)

The staff has made every effort to address in sufficient depth credible alternatives for those aspects of the decontamination which are reasonably well defined. Where there is significant uncertainty, such as the condition of the core, the most probably or bounding situations were considered. A number of options are listed for each operation and attention is focused on the most feasible options. Reasons for deletion of certain alternatives from detailed consideration are discussed in greater depth in the final PEIS. Because the time and risk associated with developing new equipment, methods has been given considerable weight in selecting procedures, the options chosen tend to use proven methods for which considerable experience exists. This fact, coupled with the generally conservative approach taken in calculating doses to workers and the public, leads the staff to believe that the depth of analysis provided is appropriate for this document. Detailed event tree analysis of all the options would significantly increase the size of the PEIS and would detract from its major objective; the overall evaluation of the environmental impacts of decontamination. If, as the work progresses, it appears that the decontamination operations actually used will differ significantly from those described in the PEIS, supplements will be issued which evaluate those operations.

13.2.1.3 Reuse of Contaminated Equipment (55, 100)

The impacts of refurbishing TMI-2 are beyond the scope of the PEIS. Equipment which is required during the cleanup would be decontaminated and restored to operation if necessary as part of the cleanup.

13.2.2 Decommissioning

13.2.2.1 Decommissioning Alternatives and Impacts (11, 16, 32, 61, 64, 66, 67, 69, 78, 84, 92, 99, 100, 114, 115, 121, 122, 124, 125)

The public has expressed strong interest in (1) the impacts of decommissioning, and (2) the effect that various decommissioning alternatives would have on the choice of cleanup alternatives. In addition an assessment of the impacts of various disposition alternatives will be needed in the foreseeable future. Therefore, an analysis of the impacts of an early decision to decommissioning has been added to the PEIS (see Sec. 2.2 and Appendix U).

13.2.2.2 Information Source on Decommissioning (32)

The staff drew upon a previous report (Technology, Safety, and Costs of Decommissioning a reference PWR, NUREG/CR0130) in its analysis of decommissioning the damaged reactor TMI-2.

13.2.2.3 Duration of Cleanup (32, 71)

Four basic factors will determine the length of time required to complete cleanup operations: the time required for individual operations; the need to carry out many of the operations in sequence rather than simultaneously; the time required to resolve differences of opinion on what should be done and how it should be done; and financial restrictions. The first two provided the basis for the initial estimate of 5 to 9 years. The third and fourth factors may increase the duration beyond this estimate. The fourth factor has been considered in the licensee's revised estimate that the duration of the cleanup will be seven years or longer.

Estimates of the times required to carry out the individual cleanup operations are based on analysis of different alternatives presented in detail in Sections 5 through 9, taking into consideration the licensee's estimates (Fig. 1.3.1). The various tasks involved in the cleanup operations must be carefully planned and executed to keep worker exposure to radiation as low as reasonably achievable (ALARA) and to ensure that the health and safety of the public are adequately protected. This approach is basic for the entire cleanup, and increases the required time over that which would be necessary for a less careful operation.

Considerable uncertainty exists in the estimates of the worker effort, occupational exposure and duration of the cleanup operations because there is very little information available on the condition of some parts of the buildings (e.g., the reactor building basement) and equipment (e.g., the fuel elements inside the reactor pressure vessel). However, it is possible to obtain reasonable bounding estimates by using knowledge of the design and operation of the equipment, well-known physical principles, the limited amount of data that is available, and experience gained in previous situations (in which similar problems occurred, even though the number and scale of problems at TMI is unique (see Sec. 1.5).

An understanding of the important factors involved and sound engineering judgement are necesssary for such estimates. One would, for example, obtain misleading results if unique or anomalous situations were used as a general basis for estimating task durations. (The long delay in opening the malfunctioning entry door is an example of an inapplicable basis.) The delay was primarily a consequence of a decision to delay further entry attempts until the NRC issued a decision on purging the Kr-85 from the reactor building.

Many of the individual estimates of upper bounds are probably conservative. For example, initial estimates of the radiation levels inside the reactor building, which were based on measurements utilizing penetrations through the reactor building exterior walls that allowed insertion of instruments at very few locations, turned out to be larger than the levels measured during the entries by a factor of 2 to 3.

The sequence of operations, and also the licensee's estimate of the duration of the operations, are summarized in Figure 1.3.1. As an example of the necessity for sequencing (and of an application of the ALARA principle), the primary coolant system cannot be cleaned up until the reactor has been defueled. Defueling will not start until decontamination of the reactor building has progressed to the point where exposure of defueling workers to radiation has been substantially reduced by decontamination operations. Decontamination cannot be started until plans based on information gained during entries have been prepared, and entries were delayed until the Kr-85 was purged from the reactor building atmosphere.

The need to resolve conflicting views on how the operations should be carried out introduces delays of variable duration that are difficult to estimate. An example is the delay in the purging of the Kr-85 from the reactor building atmosphere in order to provide time for consideration of the objections of those opposed to a direct purge to the atmosphere and to assess alternatives (see Sec. 5.2). There may be other delays in order to give full and fair consideration to different views on other issues, such as the means that should be used for disposing of the processed water. Publication of this document should help minimize further delays of this kind.

Delays due to financial constraints are also difficult to predict at this time. There must be money available to pay workers and purchase clean-up equipment and supplies. Restrictions on the amount of money available and on how it could be spent have severely curtailed the rate of cleanup since September 1980. Some of the money has and will come from insurance; the source of the balance of the money needed is not yet determined.

13.2.3 Disposal of Radioactive Waste

13.2.3.1 On Site Landfill (70)

No radioactive wastes have been or will be disposed of at the sanitary landfill at TMI.

13.2.3.2 TMI Not To Be Site for Radioactive Waste Disposal (31, 70, 125)

It is not intended that TMI be a site for the permanent disposal of either spent nuclear fuel or radioactive wastes. To use TMI as a permanent waste repository is neither compatible with current national policies nor NRC regulatory guidelines for radioactive waste disposal.

13.2.3.3 No Regulation to Prohibit Temporary Waste Storage Onsite (114)

There is no regulation that specifically prohibits the operation of a nuclear power plant with high activity radioactive wastes stored outside of the spent fuel pool. However, there are requirements that the storage of these wastes has to be safe and radiation exposures due to the storage be within regulatory limits and be maintained to as low as reasonably achievable levels.

13.2.3.4 Alternative to Dispose Processed Water by Controlled Release into River (12)

The alternative to dispose processed water by controlled release into the Susquehanna River is one of several alternatives evaluated in the statement for the disposal of processed water. Section 7.3 contains a detailed discussion on the potential environmental impacts of this alternative.

13.2.4 Current Status

13.2.4.1 Radioactivity Releases Since the Accident (60)

Since the accident, the only significant release of radioactivity was the controlled purging of Kr-85 from the reactor building. The total amount of Kr-85 released has been about 43,000 Ci. The NRC has not authorized the release of liquid radioactive waste from the site.

13.2.4.2 Reactor Building Sump Water Level (130)

Between 650,000 and 700,000 gallons of water are currently in the reactor building sump. Water level in the building increases at a small rate, on the order of an inch per month. The water level in the sump presents no immediate safety hazard. However, should transfer of this water become necessary, storage tanks onsite have sufficient capacity to accommodate this water.

13.3 TMI VICINITY AND DOWNSTREAM AREA

13.3.1 Geology

13.3.1.1 TMI Under Water (52)

Comparison of "water's edge" elevation of 280 ft. MSL on p. 3-1 of the draft PEIS with the "average elevation" of the island of 277 ft. MSL on p. 3-5 led the commentator to the conclusion that TMI is 3 ft. under water. A more careful reading indicates that the statement beginning with the last sentence on page 3-1 and continuing on page 3-5 reads "Bedrock beneath the site is essentially flat, with an approximate average elevation of +277 ft. MSL." The sentence is referring to the average surface elevation of bedrock, not ground surface. Ground surface ranges from +280 at the water's edge to more than +300 ft MSL in the north-central portion of the island, as stated in the last sentence of the next to last paragraph on page 3-1.

13.3.1.2 References on Geology (51)

The NRC is guided in its evaluation of nuclear sites by Federal Regulation 10 CFR Part 100 Appendix A to that document, which defines the seismic and geologic criteria for nuclear plant siting, and requires that in the absence of specifically identified geologic structures that have the potential of generating earthquakes in the site region, the site analysis must include a designation of tectonic province within which it must be assumed that the largest credible earthquake unique to each tectonic province can occur anywhere in that province, including its closest approach to the site. In defining tectonic provinces, the NRC staff utilizes many standard, well-accepted references including the cited, "Structural Geology of North America" by A. J. Eardley. Other references that are used routinely for this purpose by NRC Geologists include: "Seismotectonic Map of the Eastern United States" USGS MF 620 by J. B. Hadley and J. F. Devine, "The Tectonics of North America-A Discussion to Accompany the Tectonic Map of North America" Prof. Paper 628, USGS, by P. B. King; and "Tectonic Map of the U. S." USGS and AAPG, by G. V. Cohee and others. These references, although published several years ago, are still widely accepted as accurately describing the general structural geology of the eastern United States for the purpose of broad tectonic province definition. Eardley was cited because his was the main description used in the report.

The staff is not aware of any state, federal, and contracted services on TMI related studies in the areas of geology and seismology other than the site investigations by consultants to the utilities.

In regard to using "original sources," the NRC geological-seismological staff has only the capability to review data from investigations conducted by applicants for nuclear power plant licenses, information in the open literature, and when available, U. S. Geological Survey work. Other than exerting some control over the investigative methods and analysis techniques used to assess a site and conducting brief geological reconnaissances to sites, NRC geoscientists do not generate original site information.

Reference 1[1] was not used in writing Section 3.1. The geologic and seismic information generated by the utility was taken from Reference 4[2] and the published literature. Most of the data utilized from Reference 4 were results of site investigations such as topographic surveys, borings, geophysical surveys, and field geologic mapping. Except for changes in the topography during site preparation and construction, this information is still considered valid.

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[1] "Final Environmental Impact Statement Related to Operation of Three Mile Island Nuclear Station Units 1 and 2," US Atomic Energy Commission, Docket Nos. 50-289 and 50-320, 1972.

[2] "Final Safety Analysis Report, Three Mile Island Nuclear Station, Unit 2," Metropolitan Edison Co., 1974.

The staff concentrated on the upper strata at the site because the properties of that soil and rock, and the orientations, attitudes and nature of bedding, faults, and joints strongly influence the activity of groundwater. Significant faults, joints and other geologic features that are present in rocks at 1000 feet extend up through the 100 million-year-old bedrock and are detectable in the upper rock strata which has been mapped or penetrated by core borings.

13.3.1.3 Stability of the Onsite Storage Facility (75)

The NRC is guided in its evaluation of nuclear sites by Appendix A to 10 Code of Federal Regulations, Part 100, "Seismic and Geologic Siting Criteria for Nuclear Power Plants." As noted in the PEIS, all category I structures and tanks are founded in bedrock, except the Category I storage tanks, which are founded in compacted backfill. The information used to develop this input was based on information provided in the PSAR, FSAR and relevant amendments. Faults in the site area were also assessed and found to be within the meaning of Appendix A to 10 CFR 100. Based on our review of the available information in the geoscience area, we continue to conclude that the site is safe for a nuclear power plant; therefore any onsite storage facility founded in the same foundation material should be stable.

13.3.2 Meteorology

13.3.2.1 Sources of Meteorological Information (51)

Reference 9[1] for Section 3.1.3.1 is intended to provide a definite source of information regarding the TMI region's climate based on 40 years of meteorological data collection. General climatic information regarding the area could also be determined from any number of other climatology publications; however, the reference provides greater detail for the TMI region than any other generally available publication. Onsite meteorological measurements are made continuously on a tower at the north end of the island. The measurement program is described in Section 3.1.3.2. Upper air wind information is routinely available through the national weather service measurements program at Philadelphia, Pittsburgh, and Washington, D.C. This information is available on facsimile maps received by the NRC.

13.3.2.2 "Appropriate" Meteorological Conditions for Water Evaporation (99)

As described in Section 3.1.3.2, forced evaporation may produce or enhance fog in the TMI site vicinity. Evaporation of water is a function of the water temperature, the speed of the air flow across the water surface and the amount of moisture in the air over the local area. Since evaporation is most dependent on the above conditions, it is possible to limit the forced evaporation process to times when the meteorological conditions are: 1) winds 5 mph or greater, regardless of direction, 2) turbulent atmospheric conditions, 3) relatively "dry air" over the region, and 4) no precipitation. Although natural evaporation will take place from the water surface, using the "forcing" process under the four conditions given above should reduce the environmental impact of this evaporation procedure.

13.3.2.3 "Northwest Anomaly"

The statement on page 5-7, Sect. 5.1.5.2 of the draft PEIS, relative to the WNW sector at 0.37 mile from the plant was incorrect. See Appendix W of the final PEIS, particularly Table W.2, for directions and distances used for offsite close calculations.

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[1] "Local Climatological Data, Annual Summary with Comparative Data, Harrisburg, Pa," US Department of Commerce, Environmental Data Service, published annually.

13.3.3 Hydrology and River Use

13.3.3.1 Susquehanna River Hydrology (21, 79, 99, 100, 102, 130)

The minimum daily flow of record for the Susquehanna River at Harrisburg is 1600 cfs; the average flow is 34,000 cfs. Approximately 25% of the flow at the site passes the discharge point in the middle channel between TMI and Shelley Island. Below York Haven Dam, complete mixing is assumed and full flow of the river is used in determining dilution factors. Further discussion of the mixing characteristics may be found in Section 3.4 of this Statement.

The nearest potable water supply is at Brunner Island, approximately 5 miles downstream from the proposed effluent release.

13.3.3.2 Hydrology Data Corrections (35, 51)

Section 3.1.4.1 (Surface Water Hydrology) has been revised, where appropriate, to reflect the corrections and comments provided by the U. S. Geological Survey. The last two sentences on page 3-6 of the draft PEIS have also been deleted due to the recent cancellation of the Stony Creek Project.

13.3.3.3 Lowest River Flow (32, 66, 102)

The Susquehanna River has never been known to run dry. As indicated in Section 3.2.4.1, the lowest flow of record at Harrisburg was 1600 cfs on November 29, 1930, and was caused by an ice jam.

Due to the nature of hydropower operation, the flow out of Conowingo Reservoir is essentially zero at times when power is not being generated. This assurance, however, has little or no effect on downstream dilution factors and net flushing effects of the reservoir. The lack of outward flow does not stop incoming flow. With no outflow, the water can only rise upstream of the dam. The water in the reservoir has essentially the same physical characteristics as the incoming water. The water which is eventually released when the turbines are operating is the same water that originally would have gone through the turbines but was delayed a day or so by not being released.

13.3.3.4 Flood Forecast (51, 120)

The flood forecast or predicted rate of discharge for large floods will not change significantly due to the cancellation of the Stony Creek Project. The drainage area controlled by this project is minor in relation to the total drainage area of the Susquehanna River. Thus, the amount of flood control afforded by that facility is minor at TMI, and the flow rate of the river at TMI will change insignificantly.

It is true that portions of the site and the South access bridge were flooded during the Tropical Storm Agnes flood of 1972. The reason that the site was flooded was that the dikes which now protect the island were not completed at that time and water entered through the unfinished portions. The north access bridge is above the elevation of the design flood and would not be flooded by a recurrence of Tropical Storm Agnes. Section 10.5 discusses the designs of the facilities and dikes and provides pertinent elevations and flood levels.

13.3.3.5 Flood Protection (32, 67, 96, 120, 123)

Section 10.5 provides discussion of floods and flood probabilities, and of the flood protection for the island. The auxiliary and fuel handling building would not be affected by the probable maximum flood, since the building is closed off from the entry of flood waters.

13.3.3.6 Use of Susquehanna River Water for Water Supply (73, 100)

The statement on page 3-12 that "the river is not an attractive source of public water supply" has been modified. According to information provided to the NRC by the Baltimore Department of Public

Works, the city of Baltimore does not currently withdraw, and does not plan to withdraw, water from the Susquehanna River at the rate of 250 million gallons per day. The last significant withdrawal of water from the Susquehanna River for the city of Baltimore occurred in 1968 and once again in the early 1970s. The city plans to withdraw water only during prolonged drought periods when the levels in its water supply reservoirs drop very low.

13.3.3.7 River Flow Rates Considered for Dose Estimates (55, 66)

The flow that was used in the draft PEIS for estimating dose to humans due to consumption of water or fish downstream was the mean daily low flow of 1700 cfs. On the average, the flow is approximately twenty times greater than this so that it is very unlikely that controlled discharges would be made at a still lower flow. To provide more realistic assessments in the final PEIS, a river flow of 12,600 cfs has been adopted for most calculations; the basis for this number is given in Section 3.1.4.1. It should also be noted that the licensee is required to conform to the technical specifications. If the release of processed water is permitted, the technical specifications would require that the release be controlled to ensure that the public and the environment are protected from harmful doses of radiation.

13.3.4 Aquatic Ecology Studies (123)

Section 3.1.5 of the PEIS is a summary description of the ecology of the Three Mile Island vicinity and other areas potentially affected during cleanup operations. In-depth specific accounts of the aquatic ecology and fisheries of the Susquehanna River and Chesapeake Bay are included in Appendix E of the PEIS. Appendix E was prepared from an extensive review of the literature, with the most pertinent sources cited and included in the references.

No experimental research per se on the ecosystems of the TMI vicinity has been done specifically for the PEIS. However, studies of the aquatic ecology and fisheries of the Susquehanna River near TMI have been ongoing continuously since 1974. These summaries are mentioned in Section 11.10 of the PEIS. Additionally, investigations of the consequences of the accident to aquatic biota, fisheries, plants and animals of the TMI area have been completed and are contained in two NRC technical reports: NUREG-0596, "Non-radiological consequences to the aquatic biota and fisheries of the Susquehanna River from the 1979 accident at Three Mile Island Nuclear Station"; and NUREG-0738, "Investigations of reported plant and animal health effects in the Three Mile Island Areas."

13.4 REACTOR SAFETY PRIOR TO DEFUELING

13.4.1 Delays in the Decontamination Process (80)

The NRC staff recognizes the potential risks to public health and safety and to the environment from delays in decontamination of TMI-2. There are references to the effects of delay in decontamination actions in several places in the PEIS, e.g., in Sections 2.1.1.1 and 2.1.1.2, and in Appendix C. Of primary concern is the potential degradation of instrumentation and systems required for the continued safe shutdown of the reactor. The staff considers it imperative that the cleanup proceed in a timely manner, and has indicated this position to the licensee.

13.4.2 Maintenance of Fan Coolers (100, 107)

The fan coolers in the TMI-2 reactor building have operated continuously for more than 21 months without failure since the March 28, 1979 accident; eventual failure of the fan coolers can still be expected if maintenance is not performed. The controlled releases of krypton gas from the reactor building during 1980 eliminated the possibility of uncontrolled releases of the gas in the event the fan coolers failed. The urgency of maintaining them has therefore declined and is probably not necessary until some gross decontamination of these highly radioactive components has been accomplished.

13.4.3 Mini Decay Heat Removal System (32, 55)

The MDHRS is described in Section 4; it was installed as an additional method of removing decay heat from the core. However, the loss-to-ambient mode of cooling has been in effect since January 5, 1981.

The MDHRS is connected to the reactor coolant system at two points by electric-motor operated valves located in the basement. For a time there was some concern that the rising water level in the basement would cover the valve motors, incapacitate them, and prevent the valves from being opened. However, the valves have now been opened and the rising water level in the basement is no longer a threat to operation of the MDHRS.

13.4.4 Boron Analysis (32)

The comment questioned the lack of sampling of boron for several weeks last summer and the implications to maintenance of the reactor in a safe condition. The reactor is being kept in a subcritical state by the boron that has been added to the reactor cooling system (RCS) water; dilution of the boron could obviously lead to recriticality and must be avoided. To ensure that the boron concentration is adequate at all times, steps have been taken to limit sources of water to the RCS and minimize the need to make any increases and reductions in the amount of water in the RCS. Also, regular analysis of boron concentration is being performed, the frequency of which is largely dependent on the frequency of and extent to which changes in RCS water inventory are occurring.

Prior to the inspection of the RCS and defueling, there is no need to move large volumes of water in and out of the RCS. Analysis of boron concentration is then necessary less frequently and has been scheduled to be done on a weekly basis. During the summer of 1980, for a two-week period, analyses were not done by the laboratory usually doing them. However, since no large amounts of water were to be introduced into the RCS during this period, rapid dilution of boron due to accidental introduction of underborated water into the RCS was not possible. It should be noted that a neutron monitor was and is available that can detect any increases in neutron "levels" in the core that would have occurred if boron was being slowly diluted and the reactor had begun to respond to the reduced neutron absorption. The staff considers that the boron analyses were adequate and that at no time was the health and safety of the public compromised. During reactor inspection and defueling boron analyses will be done more frequently, and an additional neutron monitor will be put into use.

13.4.5 Recriticality (91, 99, 100, 115, 125)

A full description of the criticality control status of the core is given in Section 4.4.2.1. While the introduction of under borated water into the core could lead to recriticality, such an event is very unlikely and is guarded against by both engineering design and administrative controls.

To ensure subcriticality, the boron concentration in the RCS water must be maintained at more than 3000 ppm. This minimum limit was established on the basis of calculations that assumed gross redistribution of the reactor core, so that even in the unlikely event that the core melted into such a geometry, the 3000 ppm minimum limit would be sufficient to maintain subcriticality. Boron concentration in the RCS water is being maintained at more than 3500 ppm by the licensee. This limit was established on the basis of calculations that assumed the most reactive configuration of the reactor core. Much higher concentrations of boron in the water would have to occur before precipitation would present a possible problem of constricting water flow through the core; such high concentrations are improbable.

Prior to the beginning of reactor cooling system (RCS) inspection and defueling, there will be very little need for large quantities of water to be removed from and added to the RCS. Rapid boron dilution of the RCS liquid is thus all but impossible at this stage and boron sampling and analysis is necessary only on a weekly basis.

When defueling is in progress, it will be necessary to carry out some decontamination of RCS water on a regular basis because manipulation of the damaged fuel can be expected to cause release of more fission products from the damaged fuel pins into the RCS water. RCS water would then have to be processed and larger quantities of water would need to be removed from and added to the RCS. In order to ensure that under borated water is not added to the system, many procedural controls have been and would be instituted. These controls are described in Section 4.4.2.1 and they include requirements that major sources of water to the RCS be maintained at sufficiently high boron concentrations. For example, addition of water can be required to be performed in batches rather than continuously, and analysis of boron concentration can be made more frequently, perhaps as often as several times a day.

13.4.6 Dry Core (55)

The statement on page 2-8 to the effect that criticality could result from a dry core was incorrect. A dry core would not become critical, although it could become physically unstable. This has been clarified in the PEIS.

13.4.7 Fission Products from Recriticality (55)

There was a discrepancy between the summary and Chapter 4 regarding the quantity of fission products that could be released due to recriticality. The text has been clarified to indicate that a recriticality could release fission products into the reactor building but that such a release would be small and not likely to escape from the reactor building.

13.4.8 Reactor Building Sump Water Level (32, 99, 130)

Between 650,000 and 700,000 gallons of water are currently in the reactor building sump. Water level in the building increases on the order of a fraction of an inch per day. The water level in the sump currently presents no safety hazard. However, should transfer become necessary, the contingency plan calls for pumping to storage tanks which have sufficient capacity to accommodate this water.

13.5 DECONTAMINATION OF BUILDINGS AND EQUIPMENT

13.5.1 PEIS Unnecessarily Restrictive (50,75)

The staff agrees that the draft PEIS seemed too restrictive with respect to certain operations or sequences in which the decontamination operations might be performed and with respect to the use of robots for radiation measurements. The PEIS is not intended to provide detailed cleanup plans or to recommend or preclude specific methods or procedures. The responsibility for developing and implementing specific cleanup plans rests with the licensee. It is the responsibility of the NRC to review the licensee's plans and monitor the implementation of those which are approved. The analyses in the PEIS could, of course, lead to the rejection of some procedures.

The licensee's detailed plans must be prepared and then implemented in a step-by-step manner. The first steps have already been taken. They started with initial decontamination of the auxiliary and fuel handling building and processing of the water from these buildings in order to remove dissolved and suspended radionuclides. Subsequent steps included construction of facilities for packaging and temporary storage of radioactive waste, and preparation for cleanup of the reactor building and reactor by purging the Kr-85 from the reactor building atmosphere and surveying the radiation levels and damage inside the building. The foregoing activities proceeded, with appropriate review and monitoring by the NRC, prior to publication of the PEIS. The next steps will be to prepare and then implement detailed plans for decontaminating the reactor building, defueling, cleaning up the primary system, and disposing of the wastes, including the processed water. These plans will be prepared by the licensee and then reviewed and monitored by the NRC, taking into consideration the guidelines provided by the PEIS.

13.5.2 Radioactivity within the Buildings (50, 107)

Early estimates of the amount of radioactivity in the reactor building and the auxiliary building were based on remote sensors and indirect measurements. However, several entries have been made into the reactor building and the actual measurements show that most of the radiation levels are less than the original estimates. Also, a large portion of the cleanup in the auxiliary building is already completed.

13.5.3 Personnel Training (79)

Although the draft PEIS makes no specific mention of training, the decontamination effort will require extensive training of all types of personnel involved in the cleanup operations. Maintenance personnel, equipment operators, engineers, reactor operators, and health physics will be included. Formal lecture and discussion programs dealing with the radiation hazards and practical approaches to keep exposures as low as reasonably achievable (ALARA) have already been instituted by the licensee. These programs are designed to prepare cleanup personnel for the situations they will encounter. Demonstrations, trial dry runs, and work on mock-ups of reactor systems will also be needed in order to maximize the efficiency of the workers and minimize the time spent in the radiation fields. Planning sessions will be utilized to develop the best way to perform each task and to employ experience gained in previous tasks to best advantage

In addition, certain decontamination tasks will require specialists with proven experience to perform them. Outside organizations can and have been contracted by the licensee for some of these specialized jobs.

13.5.4 Condition of Equipment in the AFHB (100)

The condition of the equipment in the Auxiliary and Fuel Handling Building has been questioned, particularly whether this equipment will meet safety requirements during the cleanup.

The Auxiliary and Fuel Handling Buildings and their equipment were designed to meet all applicable safety regulations. They were also designed to prevent radioactive material released to the environment from exceeding the limits established by applicable regulations. The equipment damage is primarily due to flooding and atmospheric conditions which caused widespread contamination. After the buildings and equipment have been decontaminated, and repaired, they should be as safe as before the accident. The NRC wil inspect them in order to make sure that this is actually the case.

13.5.5 Concrete Surface Removal (71)

Most of the radioactive contamination deposited on building surfaces can be removed by rinsing, washing with a high-pressure water-jet, or scrubbing with detergent. However, in some cases the radioactive contamination has diffused into the surface and a surface layer of concrete must be removed in order to get rid of it.

A prior situation involving an accident at a reactor in which concrete removal was necessary for decontamination occurred at Chalk River in Canada.[1,2] The Chalk River experience involved removal of concrete "down to the aggregate." Then concrete was added to a depth of about 6 inches greater than the original wall to act as additional shielding.

At TMI-2, most of the concrete surfaces were coated with materials to limit the absorption of liquid-containing contaminants. However, concrete removal may be required in two areas. Below the 305-ft level in the reactor building, the walls were coated only up to about 5-1/2 feet above the floor. The sump water has been in contact with about 2-1/2 feet of noncoated, but very dense, concrete. As a result, surface removal is anticipated down to a depth of about 0.5 inch. This estimate is based on the experience in the AFHB, where sump water was in contact with noncoated concrete for about a year. The maximum surface removal required was 3/16 inch.

The second area that may require some concrete removal is the elevator shaft in the reactor building. The concrete blocks of the elevator shaft were not coated and some surface removal is anticipated. The blocks will be cored prior to decontamination to determine how far the contamination may have penetrated.

13.5.6 Alternatives for Removing the Sump Water from the Reactor Building (50, 75)

The list of sump-water removal alternatives has been expanded in the final PEIS (see Sec. 5.2.2.1). The removal procedures involve a number of tasks that must be coordinated with other cleanup activities. There are many alternatives for each task and coordination possibility. The total number of overall alternatives is very large, making it impossible to carry through a detailed analysis and selection of an optimum plan prior to the start of cleanup. This would be true even if the occupational dose could be accurately predicted for each task.

Many uncertainties (including radiation sources remaining in the basement after the sump water has been pumped out or the amount of water that will be needed for decontaminating the reactor building) cannot be resolved until cleanup operations have progressed to where detailed, accurate data become available. At the present time, we can only make approximate bounding estimates of occupational exposures and environmental impacts. These are based on reasonable assumptions for a sampling of alternatives. An overall approach can be sketched out; but detailed plans and decisions regarding individual task alternatives must be made on a task-by-task basis.

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[1] Harry Collins, John Logie; "Canadian Decontamination Experience:" Chalk River Nuclear Lab.

[2] J. W. Logie; "Three Vessel Replacements at Chalk River;" Atomic Energy of Canada Limited; Decontamination and Decommissioning of Nuclear Facilities, Marilyn M. Osterhout, ed.; 1980.

13.5.7 Installation of Large Scale Cleanup Systems (20)

Decontamination and cleanup systems in operating nuclear power plants are designed and licensed for safe operation of the facility and reduction of radioactivity releases during normal operation and anticipated transients. These systems are not intended for the large-scale cleanup operation necessary following a major accident. Installation of large-scale cleanup systems, like those considered for the post accident cleanup at TMI-2, will not improve the safety of normal operation nor will they significantly reduce normal releases from operating reactors.

13.5.8 Effects of Prior Spill (32)

The spill "prior to the accident" mentioned on page 5-25 of the draft PEIS consisted of minor amounts of resin from flush bags used during the servicing of the condensate polishing unit and flushing of the demineralized water system. Reportedly, less than 5 gallons of resin were involved. The incident is recorded in the TIMI-operation records. Flushing the small amount of spilled resin into the sump is an effective "housekeeping" measure and is representative of the type of approved housekeeping that takes place in nuclear power plants. The extent to which the small amounts of resin in the auxiliary and fuel handling building complicated the accident cleanup is negligible. The changed description of the spill in the final PEIS will better reflect its true importance.

13.6 REACTOR DEFUELING AND PRIMARY SYSTEM CLEANUP

13.6.1 Uncertainties in Plant Conditions

13.6.1.1 Best-case/Worst-case Analysis (75)

The staff has stated in the draft PEIS (Sec. 7.1) that because of unknown conditions within the reactor pressure vessel and because of the difficulty in performing an accurate, detailed analysis of vessel conditions, the condition of these components will not be known until inspections and examinations can be performed. For these reasons, the staff has presented an evaluation in the document of the impacts associated with best-case/worst-case conditions. Actual conditions should be within those bounding conditions presented.

The best-case/worst-case approach to TMI-2 cleanup activities was applied to provide limiting conditions. The purpose of the analysis is not to specify cleanup activities for the licensee but to provide an environmental review of the several bounding alternatives. The weakness of such an approach is that it does not help in predicting realistic budgets.

Several studies have been performed concerning the accident which, based on conditions experienced during the accident, describe possible reactor building conditions. "Analysis of Three Mile Island-Unit 2," a NSAC report, and "Three Mile Island, a Report to the Commissioners and to the Public," by the Special Inquiry Group of the U. S. Nuclear Regulatory Commission, were used extensively to establish the most-probable and worst-case conditions that were used in the draft PEIS. The data in these reports indicate that there will not be any overwhelming problems for health and safety arising from the uncertain conditions in the reactor pressure vessel or in the core.

13.6.1.2 Need for New Technological Developments (4)

The staff does not agree with the assertion that "safe operations will require new technological developments that are beyond the present state of the art." As stated in the PEIS (Chapter 12 and elsewhere), the staff concluded from its analyses of the cleanup that existing methods are adequate, or can be suitably modified, to perform all of the necessary operations with only minimal releases of radioactivity.

13.6.1.3 Effects of Unknown Reactor Conditions on River Water (13)

Concerning the effect of unknown reactor core conditions on effluents that may be released to the river, it must be stressed that release of liquid effluents (after dilution) is only one of the disposal alternatives considered in this document. The accident water in the AFHB and RB and the existing primary coolant have been analyzed for radionuclide content (Sect. 7.1); these liquids represent the greatest part of the radioactivity to be processed. The condition of the core will not affect these values.

The core condition will affect the RCS decontamination and flushing liquids. Here the uncertainty is much greater, but under the worst-case conditions the total radioactivity sources during the RCS decontamination and defueling will be a small fraction of the sump inventory. Further, a significant part of the activity released may be in the form of suspended particulates that will be removed in the first filter stage of any processing alternative and disposed of as solid waste.

Tritium, however, cannot be so removed. Tritium is formed almost entirely in the primary coolant water during reactor operation; very little tritium is formed in the fuel at the present time. Thus, the condition of the core will not significantly affect the tritium content of the water to be processed or the tritium concentration in the effluents.

13.6.1.4 How Much Kr-85? Where Is It? (50, 55, 72)

The total inventory of Kr-85 just before the accident was 100,000 Ci, as determined essentially from reactor burnup information. Its location was all in the uranium oxide pellets inside the zirconium cladding of the fuel rods. After the accident, about 45,000 Ci was trapped in the reactor building and subsequently was vented to the atmosphere. The remaining 55,000 Ci can be divided into (a) a portion still in the fuel, (b) a solute in the primary cooling water, and (c) releases to the atmosphere during the accident. The solubility of krypton in water, especially when the water is at high temperature, is small enough so that even 700,000 gallons would not hold a significant inventory. The nuclide Cs-137 is known to have about 60% of its inventory outside the fuel elements. This suggests that some 40,000 Ci of Kr-85 remains in the fuel, but it might be less.

13.6.1.5 1.5 Curie of Kr-85 per Fuel Element? (55)

The 45,000 Ci of Kr-85 in the fuel is divided into about 30,000 fuel pins, or 1.5 Ci per pin, not per fuel element (there are about 200 fuel pins per element). This is an error in the draft PEIS, page 8-13, and has been corrected.

13.6.1.6 Leach of Radionuclides from Spent Fuel Rods (73)

It has been suggested that all radionuclides in the spent fuel may leach out through the destroyed cladding into the primary coolant water. Radiochemical analysis of the primary coolant water has been conducted weekly since shortly after the accident and no further radionuclides have been found in the primary system water. The staff cannot be absolutely sure that significant leaching will not occur; therefore, the alternatives in the final PEIS include cases where there is little or no leaching, and cases where significant leaching occurs.

13.6.2 Reactor Vessel

13.6.2.1 Reactor Vessel Inspection (75)

Reactor inspection consists mainly of examination of the reactor core and internals to assess the damage and plan for cleanup. Recent entries into the reactor building were conducted mainly for radiation surveys and assessments of contamination. No significant reactor inspection work has yet been performed.

13.6.2.2 Reactor Vessel Integrity (50, 100)

The exact structural integrity of the reactor vessel is unknown at this time. We do know positively, however, that the reactor vessel integrity is completely satisfactory for current and contemplated future cleanup operations. The facts are: after the core severely dried out during the accident, pressure in the pressure vessel was repeatedly cycled from approximately 1200 to 1100 psi. During these pressure transients, there was no indication of loss of integrity. The reactor pressure vessel is now operating at less than 100 psi and will continue to be maintained at this pressure or lower until after removal of the fuel. Because the reactor vessel is essentially new (very little neutron exposure), nulductility problems do not now exist nor will they in the foreseeable future.

The condition of the reactor vessel is pertinent to cleanup activities only to the point that the core maintains its integrity to hold coolant water. No cleanup activities are envisioned that would require that the core be subjected to conditions beyond those experienced so far. Structural conditions would require evaluation prior to recommissioning, but this is not within the scope of the PEIS.

13.6.2.3 Radiation Levels during Defueling (50)

The licensee stated that the staff was unrealistic in saying the residual reactivity of the reactor building (after decontamination of the reactor sump water and hot spot shielding) would make no contribution to the radiation levels around the defueling area. Based upon radiation levels measured during the entry on July 23, 1980, the text has been modified, in Appendix I of the PEIS, to reflect a value of 5 mR/hr, which is the criteria of the initial building decontamination prior to defueling and is felt to be readily achievable.

13.6.2.4 Unsuitable Vessel Diagram (50)

Figure 8.1.1 is not precisely applicable to TMI-2. However, it is a typical representation of current generation LWR(S). The caption has been changed.

13.6.2.5 Contamination of Reactor Vessel Surfaces (32)

A statement in Section 5.1.4.1 that the distribution of different nuclides in the surface contamination would be similar to the distribution in the reactor building sump water was criticized. Although the surface contamination came from the water now in the sump, we agree that different rates of precipitation and settling for different elements would change the relative concentrations of different elements, and therefore nuclides. What is important is that measurement must be made during cleanup to guide the detailed choices required during that step-by-step procedure.

13.6.3 Working Time and Productivity Estimates

13.6.3.1 Criteria for Estimation of Working Time (50)

The licensee suggests that the best-case working time estimate described in Section 8.1.3 was conservative when compared to "normal" conditions, and that some operations could be carried out with less working time than estimated by the staff.

Several studies have been performed concerning the accident which, based on conditions experienced during the accident, discuss possible reactor building conditions. "Analysis of Three Mile Island-Unit 2 Accident," a NSAC report and "Three Mile Island, A Report to the Commissioners and to the Public," by the Special Inquiry Group of the U. S. Nuclear Regulatory Commission, were extensively used to establish the best- and worst-case conditions which have been applied in the PEIS. Because of the data in those reports, the staff's judgement was that even in the best-case, "normal" conditions as known in the past will not exist in the activities involved in the reactor defueling.

13.6.3.2 Time Estimate for Core Removal (50)

It was suggested that the staff's estimate of 10 months for head and plenum removal and defueling is too conservative, even for worst-case conditions. The NRC staff believes that worst-case conditions are rather probable, so that the hope that this part of the cleanup can indeed be done in less than 12 months is unrealistic.

13.6.3.3 Worker Productivity (50, 75)

The "productivity factor" is used to take into account the decrease in efficiency of personnel working in contaminated areas. This factor (expressed in percent) gives the reduction in the amount of work accomplished per unit time due to encumbrances from protective clothing, face masks, respirators, etc. Factors in the range of 40 to 67% were used for different tasks and situations. These factors apply only to the time spent working in contaminated areas; they do not include time spent in putting on and removing protective clothing, planning, training, receiving instructions, etc. The fraction of time that decontamination personnel spend working in contaminated areas ranges from 30 to 60%; the staff assumed 50% as an average figure for decontamination work. The remaining time would be spent in the preparatory activities noted above, and may be regarded as part of the overhead. (The remainder of the overhead would be for support workers with assignments that do not require entry into contaminated areas.) The numbers used for productivity factors and "preparatory overhead" are based on experience in the industry for similar decontamination and cleanup activities. They represent the best estimates that can be provided at the present time of worker efficiency in environments and under conditions that, for many tasks, have yet to be determined.

13.6.3.4 Workers Volunteer for Cleanup Tasks (46)

Workers for the cleanup are volunteers in that they are informed of the natue of the cleanup tasks and have chosen to undertake the assignments. These cleanup workers are provided with appropriate training including information on the potential health effects of radiation exposure. They are not being paid extra for particular cleanup functions; i.e., workers are paid on the same scale for the same work functions at TMI, whether the work is related to cleanup or not.

13.6.3.5 Extrapolation of Data from Work Already Done (32)

As indicated in Section 5, information from the work effort already expended in decontaminating the AFHB has been utilized in making estimates for the cleanup work yet to be done on building and equipment surfaces and processing contaminated water. This experience has been applied in the staff's estimates regarding decontamination work in the reactor building, and indicates that both time and person-rem can be saved through the use of careful planning. Unfortunately, the removal of the reactor vessel head and subsequent activities including removal of the fuel and debris involve different operations than simple building decontamination. Even so, it is obvious that some experience from the building decontamination efforts will be beneficial to the defueling operations.

13.6.4 Need for Special Tools (20, 125)

In Section 8.2.2.2 of the draft PEIS, the staff mentioned the clamshell as an idea for fuel removal under worst-case conditions. Although this approach is not very probable, it is a possible alternative. Regardless of the defueling means employed, the possibility of releasing krypton gas exists.

Section 8.2.3.3 provides a list of special tools and equipment that may be needed. Specific tool requirements are not entirely identifiable at this time due to lack of detailed knowledge of core conditions. However, engineering capability exists to develop whatever tools may not be available.

13.6.5 "Corridor Concept" (100)

The discussion of this concept in Section 2.1.2.2 of the draft PEIS indicates the need for a considerable amount of space for laydown areas and the movement of fuel handling machines, etc. This is the reason why the staff does not regard the corridor concept as workable.

13.7 WATER PROCESSING AND PROCESSED WATER DISPOSAL

13.7.1 Water Processing

13.7.1.1 Airborne Releases of Radioactive Material from Water Processing (32, 55, 70, 73, 100, 116, 120)

In order to estimate airborne releases of radioactive material from water processing, a distinction between the behavior of tritium as a fission product as opposed to most others present in the various waters to be processed should be noted. Tritium is an isosope of hydrogen and it combines with oxygen and hydrogen to form a molecule (HTO) with the same chemical properties as water. When water is evaporated, all water molecules, including those with tritium (HTO), exist in a vapor state and would pass through a filter system uncollected. Other fission products (and actinides, such as plutonium or uranium) exist in the process water as dissolved constituents. When the water is evaporated, the dissolved constituents stay behind in the remaining liquid. Evaporation is thus an effective method for separating bulk water from a solution containing dissolved radioactive constituents, but is totally ineffective for separating tritium.

The action of evaporation also creates turbulence at the surface which results in the formation of small droplets of water that are physically entrained in the rising vapor. The vapor contains only tritium, no dissolved radioactive materials; the entrained droplets contain both tritium and dissolved radioactive materials. In a well-designed evaporator, the fraction of the total material that exists in the form of entrained liquid droplets seldom exceeds 1×10^{-5} of that in the original liquid. This then forms the basis for estimating how much dissolved radioactivity appears in the vapor stream. Following condensation, most of the entrained radioactivity appears in the condensate with a small quantity remaining with the uncondensed vapors. About 1×10^{-3} of the radioactivity in the condensate remains in the uncondensed vapors. Thus, less than 1×10^{-8} of the initial radioactive material except tritium remains in uncondensed vapor.

Other processing operations are generally less energetic and less turbulent than evaporation. In ion exchange operations, process liquids are pumped in closed pipes through an ion exchange column. The fluids are relatively free of substantial turbulence. However, even in these operations, a fraction of the process liquids may form small droplets carrying with it the dissolved radioactive material and that are physically entrained into surrounding air spaces, and subsequently transported by the process off-gas system to the air cleaning system. As some of the liquid droplets evaporates en-route, the dissolved radioactivity is left as a solid aerosol in the air stream. Entrainment for this kind of operation seldom exceeds a fraction of 1×10^{-6} of the process liquids.

For the purposes of estimating airborne releases for the processing of TMI liquids, it was assumed that a fraction of 1×10^{-4} of the processed liquid becomes airborne and enters the process off-gas streams. This value is conservative when compared with the information presented above and is compatible with data obtained from the processing of nuclear fuels. The air cleaning system for the removal of particulate matter consists of the HEPA filter. Regulatory Guide 1.140 gives guidance to the testing procedure and requirements for installed HEPA filters. The required efficiency for each stage tested is 99.95%, which corresponds to a penetration fraction of 5×10^{-4}; the penetration through two stages of HEPA filters (properly tested and qualified) would be $(5 \times 10^{-4})^2$ or 2.5×10^{-7}. In this statement, a penetration fraction of 1×10^{-3} was used for calculating effluents and releases for a filter train of two HEPA filters in series, a value that is very conservative to achievable values.

The failure of a HEPA filter was considered as a credible accident in this statement. The releases of the facility are constantly monitored with appropriate instrumentation. A failure of a HEPA filter would be evidenced by an increase in releases by the monitoring system and would alert operating personnel to take corrective action such as securing the ventilation system. The failure would likely occur only in one of the HEPA filters in the filter train, thus, containment integrity for the system would be maintained but at a reduced level. The detection of a significant failure within 15 minutes of its occurrence is considered reasonable. No failures of the HEPA filters occured during EPICOR-II processing of contaminated waters.

13.7.1.2 Processing of Accident Water in AFHB (11, 72, 74)

Soon after the accident, it was recognized that the containment water in the AFHB would pose an imminent problem and had to be expeditiously cleaned up. The approval of the EPICOR-II System, however, received extensive review by the NRC staff and the public commented on the environmental assessment issued for the evaluation. The licensee is evaluating methods for solidification of the EPICOR-II resins. The solidified wastes will probably be shipped to a DOE waste disposal facility. The DOE is currently studying suitable sites for disposal.

13.7.1.3 Processing of Accident Water from Reactor Building (27, 28)

The reactor building sump (RBS) water differs from the auxiliary and fuel handling building (AFHB) water chiefly in having about double the volume and ten times the total initial radioactivity. Because of the higher radioactivity concentration, the RBS water may be processed by a system using the Submerged Demineralized System (SDS), if approved. The activity level of the processed RBS water will be of the same order of magnitude as that of the processed AFHB water and the same disposal options will be considered (as discussed in Sec. 7).

13.7.1.4 Processing System Optimization (75, 81)

Optimization of processing systems for TMI liquids is not within the scope of this PEIS; rather, the discussion here considers existing state-of-the-art water-processing methods and evaluates the degrees of decontamination that are achievable through their use either alone or in combinations. Design optimization would occur at a later date when specific decontamination processes are chosen for application at TMI. This statement does, however, utilize revised decontamination factors for SDS and evaporators that are based on more recent laboratory or operational experience.

13.7.1.5 Chemical Interference with Water Processing (21, 79, 99)

Regarding materials which may interfere with water processing, oil and grease will be removed as far as possible prior to processing, and residual amounts will be trapped in prefilters. In considering processing alternatives, solutions containing detergents or chelating agents, and particularly those containing more agressive chemical reagents, are regarded as separate processing streams, distinct from the major water sources (accident water, primary coolant, RCS decontamination and flushing liquids). The decontamination of these streams is discussed in detail in Appendix G.

13.7.1.6 State-of-the-Art Decontamination and Volume Reduction Factors (50)

Although state-of-the-art radwaste processing equipment can provide improved decontamination factors (DF) and volume reduction factors, those discussed in Section 7.1 and Appendix G were selected based on operating experience at nuclear facilities. These factors also provide a measure of conservatism to account for abnormal operating conditions. The selection of these parameters provide realistic bounding estimates for processing liquid wastes.

13.7.1.7 Use of Other Ion Exchange Materials (26)

The ion-exchange systems selected for discussion in this statement have been proven, through laboratory and field applications, to be applicable for the specific radiochemical conditions and volumes of liquid to be processed. Should other proven systems become available that provide significant operational, impact, and/or cost improvements over existing systems, they will be considered if they are proposed by the licensee.

13.7.1.8 Processing of Decontamination Liquids (50)

Ion exchange techniques, among other options, are being considered for processing of decontamination liquids. In general, however, decontamination liquids have detergents and other cleaning chemicals that are not compatible with ion-exchange systems. These processing alternatives are discussed in Section 7.1 of the PEIS. Corresponding flow diagrams are provided in Appendix G.

13.7.1.9 Consistency of Inventory of Processed Water (79, 85)

The volumes and concentrations of radionuclides in the processed water are dependent on the volume and concentration of the input streams and the water processing system. These parameters establish the bounds for the processed water inventory for the best and worst cases. They are discussed in Section 7.1.

13.7.1.10 EPICOR-II System Uses (50, 55, 115)

Variations of the EPICOR-II and the Submerged Demineralizer System (SDS) have been considered for both Reactor Building (RB) sump and primary system liquid processing, e.g., the EPICOR-II System is considered for polishing RB sump liquids in Section 7.1.

13.7.1.11 EPICOR-I System Uses (55)

As for whether the EPICOR-I system would require an environmental assessment if used for Reactor Coolant System (RCS) water cleanup, it should be noted that the EPICOR-I system is used for low-level contaminated water (<1 μCi/mL) and is unsuitable for processing primary system (RCS) water. There are no plans to use EPICOR-I for TMI-2 decontamination and the system was not evaluated in the PEIS.

13.7.1.12 Proprietary Information on EPICOR-II System (84)

A "Confidential Disclosure Agreement" exists between the NRC and EPICOR, Inc. regarding the disclosures of proprietary information to enable the NRC staff to review the safety of the EPICOR-II System.

13.7.1.13 Processing of Dissolved Gases in Liquids (55)

Process equipment is designed with vent lines which are directed to the plant ventilation systems. These ventilation systems are monitored prior to discharge to assure releases meet release limits. In the evaluation of liquid processing alternatives, considerations were given to the effluents of dissolved gases. However, it is expected that the quantities of dissolved gases which could be released during liquid processing would not be significant.

13.7.1.14 Reactor Building Sump Liquid Processing Systems (50, 81)

Current design considerations and decontamination factor (DF) for the zeolite/resin systems to be used for processing Reactor Building Sump liquids are discussed in Section 7.1. Possible modifications to process parameters needed to achieve the design DF's are also taken into account.

13.7.1.15 Reactor Coolant System Liquid Waste Estimates (32)

The value of 200,000 gallons maximum of liquid waste generated in decontamination of the reactor coolant system was in fact obtained from experience at the Dresden decontamination operations.

13.7.1.16 EPICOR-II and Boron Removal (50)

The interaction between the boron content of primary water and the action on ion-exchange system, EPICOR-II, has been taken into consideration. It has been assumed that using EPICOR-II to remove radioactive ions from the water would also remove the boron needed to keep the core subcritical. In fact, EPICOR-II can, and has, been used in such a way that leaves the boron in solution, and this consideration is factored into the discussion of this statement.

13.7.1.17 Usage of RCS Makeup Purification System (50)

The RCS makeup and purification system could be used if required. However, other factors have to be evaluated as cleanup progresses, for example the condition of the system integrity such that it would not become a source of reactor coolant leakage. The staff does expect that this system will be cleaned up and that the basic piping, valves, pumps, and other components can be used to handle the RCS water before its usage can be considered.

13.7.2 Reuse and Disposal of Processed Water

13.7.2.1 Disposal Alternatives for Processed Water (13, 27, 28, 31, 55, 66, 67, 72, 75, 76, 79, 84, 85, 91, 100, 112, 126, 130)

The alternatives for disposal of processed water is discussed in detail in Section 7.2. Section 7.2 provides detailed assessments on the alternative for the disposal of the processed water, including facilities, effluents and environmental impacts under normal operation and under accident conditions. While the actual method of disposal has not been selected, the proposed method of processed water disposal to be submitted by the licensee will be evaluated based on the alternatives discussed.

13.7.2.2 Radioactivity Levels in Water for Processed Water Release Alternative (12)

At present, the NRC has not authorized the release of processed accident-2 contaminated water from TMI. The PEIS does contain an environmental assessment of the alternative for the controlled release of processed accident-2 contaminated water into the Susquehanna River. If the NRC decides to authorize the release, the NRC will act to ensure that the licensee will implement, according to the conditions (technical specifications) and criteria determined by the NRC, to ensure the health and safety of the public including people who may drink the water and consume the seafood. It should be noted that the PEIS concludes that the concentration of radioactivity in the water at the nearest drinking water intake downstream of TMI will be below the EPA drinking water standards (which must be met by the operators of the drinking water distribution systems) and safe for drinking during any controlled release of the processed water from TMI. The PEIS also concludes that the absorbed radiation dose in fish and seafood in the river and Cheasapeake Bay area would be an insignificant fraction of the normally occurring natural background radiation dose and should cause no detectable biological effect. Consequently, consumption of fish and seafood should cause no harmful health effect because of radioactivity contamination.

13.7.2.3 Water Discharged and EPA and State Requirements (20, 67, 114)

The present Clean Water Act does not prohibit discharge of low level radioactive waste into navigable rivers. The prohibition refers to the discharge of radiological warfare agents or high-level radioactive waste.[1] Low level waste discharge limits are either promulgated by an individual state (agreement states) or are governed by NRC regulations (nonagreement states). Effluent limits from commercial reactors, however, are under the jurisdiction of the NRC.

The staff will evaluate the impacts of any discharge, if authorized, at the level permitted by EPA or the state, as appropriate. Actual compliance with the Clean Water Act and its implementing regulations is subject to the jurisdiction of EPA and/or an authorized permitting state.

13.7.2.4 Long-term On-site Storage Would Require Special License (52)

Unless the solidified processed water was determined to be a "nonradioactive" waste, licensing of TMI as a disposal site would be required for long-term storage of wastes. A disposal license is not required for waste storage.

13.7.2.5 Solidification and Storage of Processed Water (50, 55, 72, 76, 84, 114)

It is not intended that the TMI site to become a de facto long-term depository for radioactive waste. Intentional temporary storage of processed water and some solid wastes is inevitable, prior to approval of methods and sites for their disposal.

Solidification of the processed water would not reduce the volume of the water or the mass and volume of the resultant bulk. Prior processing of water to reduce the radioactivity level would still leave tritium content undiminished. Although concrete made with decontaminated water would present an insignificant radiation hazard, there is currently no definition of a lower limit for the radioactive concentration of waste regarded as radioactive and subject to regulation. Technically, therefore, storage of this concrete would make the island a low-level waste disposal site.

Vitrification, which is essentially the encapsulation in a glass media, is not feasible for the water per se. However, vitrification of the resins and ziolites is a possible alternative discussed in the Sections 8.1.2.2 and 8.1.2.3.

13.7.2.6 Direct Solidification of Water in Reactor Building Water (50, 52, 72)

Alternatives considered for cleanup of reactor building and primary system accident water are discussed in Section 7.1. One of the alternatives considered is direct immobilization of unprocessed accident water. This scenario is summarized in Section 7.1. For a number of reasons, including implementation time and occupational exposures direct solidification of unprocessed accident water is considered impractical.

13.7.2.7 Methods for Concentrating Tritium (72)

There is no large-scale tritium-concentration technique for the amount present in the more than one million gallons of wastewater at TMI-2.

13.7.2.8 Reuse of Processed Accident Water (102, 126)

Processed water availability, reuse applications, and limitations are identified and discussed in Appendix F.

13.7.2.9 Dilution of Processed Water During Disposal (58)

The feasibility of using low tritium concentration water to dilute the processed water such that the tritium concentration of the diluted process water would be below that of the river water is not likely since the tritium levels in the river upstream of the plant are significantly different than those of other nearby water sources. For example, the concentration upstream in the river is 1.8×10^{-7} µCi/mL and even if the low tritium source had 90% less tritium (1.8×10^{-8} µCi/mL), it would take about 13×10^{6} acre-feet of water to dilute 1000 Ci of tritium to river concentration.

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[1] Section 301 subparagraph F, "The Clean Water Act showing changes made by the 1977 Amendments," Serial No. 95-12, U. S. Government Printing Office, Washington, 1977.

13.8 SOLID WASTE MANAGEMENT

13.8.1 NRC Regulations and Criteria

13.8.1.1 Definitions of Low-Level and High-Level Wastes (64, 70)

High-level radioactive wastes have been defined in 10 CFR 60 as "those aqueous wastes resulting from the operation of the first cycle solvent extraction system, or equivalent, and the concentrated wastes from subsequent extraction cycles, or equivalent, in a facility for reprocessing irradiated reactor fuels." Irradiated nuclear fuel is also considered to be high-level waste. Low-level wastes are considered to be all other wastes not defined as high-level wastes. Based on these definitions, EPICOR-II wastes are considered to be low-level waste even though they will require shielding for handling and shipment.

Within the category of low-level wastes, various disposal approaches may be warranted; e.g. some of the high-specific-activity wastes resulting from TMI cleanup will not be acceptable for routine shallow land burial. For those wastes which are unacceptable for routine shallow land burial, a case-by-case evaluation will be performed to determine which offsite treatment and/or disposal methods will be required.

13.8.1.2 Waste Management Regulations (32)

Regulations regarding the disposal of low-level radioactive wastes are currently in 10 CFR 20. However, 10 CFR 20 provides only broad requirements for waste disposal and does not contain specific requirements for such wastes as those which may be generated during the TMI cleanup.

The NRC is currently preparing specific regulations for high-level and low-level waste management in 10 CFR 60 and 10 CFR 61, respectively. The procedural part of 10 CFR 60, the high-level waste regulation, was published in the Federal Register as a proposed rule on December 6, 1979. The technical part of 10 CFR 60 was published as an Advanced Notice of Proposed Rulemaking in the Federal Register on May 13, 1980. A proposed rule is scheduled to be published in late 1980. The high-level waste management regulation is scheduled to be published as a final rule in end of 1981. The low-level waste management regulation, 10 CFR 61, is available for public comment as a preliminary draft. A notice of availability of 10 CFR 61 was published in the Federal Register on February 28, 1980.

Disposal of wastes generated at TMI will be carried out consistent with these regulations even if final rulemaking proceedings may not have been completed.

13.8.1.3 Standards for Waste Disposal (50, 100)

The final waste forms and contents of the wastes generated to date (including EPICOR-II wastes) are not yet completely established by the licensee. However, the NRC has stated that wastes which have the same characteristics as those routinely generated from nuclear power plants would be acceptable for disposal at commercial shallow land burial sites. In addition, for those wastes which are clearly unique to TMI (e.g., first stage EPICOR-II wastes), NRC has provided qualitative disposal criteria. The NRC staff has indicated that high-specific-activity EPICOR-II ion-exchange wastes could be disposed of at a commercial land burial site at an arid location if these wastes are solidified, or placed in a high integrity container, and special disposal procedures are included to minimize exposure to inadvertent intruders. However, the most practical alternative is to transfer these wastes to an existing federal government facility for future processing and eventual disposal. In addition, NRC staff has also indicated that any first stage zeolite wastes from the proposed Submerged Demineralizer System (SDS) having Cs-137 activities on the order of 1000 Ci/ft^3 from processing reactor building sump liquids are more like high-level wastes than material that is normally disposed of at commercial land burial sites and the assumption that these wastes can be disposed of in a commercial shallow land burial system is not valid.

For unique wastes which have not been encountered in the commercial nuclear industry and are yet to be identified, specific disposal criteria cannot be developed since the criteria would depend on the characteristics of radionuclides present in the wastes, e.g., the specific activity, physical and chemical form of the waste product as well as the available disposal options. NRC will assign a high priority to making decisions regarding the management of wastes following the receipt of data which provide the characteristics of the proposed wastes and the proposed disposal or storage options.

13.8.1.4 Case-by-Case Evaluation of Unique, Non-Routinely Generated Wastes (22, 32, 50, 53, 76, 100)

Some of the waste forms which will be generated during the TMI-2 cleanup will have different characteristics than wastes routinely generated and disposed of at commercial shallow land burial facilities. These special wastes include first stage EPICOR-II resins which have specific activities of approximately 40 Ci/ft^3. Because of their characteristics, these materials cannot be disposed of using routine methods at low-level disposal facilities. The most practicable alternative is to transfer these wastes to an existing federal government facility for future processing and eventual disposal. Other less desirable alternatives include processing on-site and/or packaging in high-integrity containers for special disposal at commercial LLW burial sites. Evaluations of waste characteristics will be performed by the NRC after the licensee identifies that a special waste form will be generated but prior to generation of the special form. Organic ion-exchange material will be limited to 10 Ci/ft^3. The evaluations will include considerations of the commercial and non-commercial disposal options available for the wastes, transportation modes and on- site and off-site storage provisions. Confirmatory testing by NRC consultants will be performed where necessary to verify licensee-submitted data.

Physical, chemical, and radiologic conditions in wastes can be simulated by using nonradioactive materials, tracer isotopes, gamma irradiators and research reactors. These simulated conditions can be used to obtain realistic data on the characteristics of wastes to be generated at TMI.

The performance objectives for waste disposal evaluations will be (1) doses to inadvertent disposal site intruders will not exceed 500 mrem/yr and (2) doses from groundwater pathways will not exceed 25 mrem/yr at the disposal site boundary. For transportation, the criteria are specified in the DOT regulations 49 CFR 171 to 179 and the NRC regulations 10 CFR 71. For storage of wastes personnel exposures will be as low as reasonably achievable (ALARA) and migration pathways into the environment will be minimized under both normal and accident conditions.

13.8.1.5 Criteria for Waste Packaging (75)

The choice of packaging for the wastes generated at TMI is dependent on the physical and chemical characteristics, the specific radioactivities and radiation levels of the wastes. NRC regulations govern the choice of packaging and extent of shielding specified during the packaging, handling, storage, and transportation operations. DOT regulations govern shipping mode and route selection. The regulatory constraints and feasible alternatives for these waste management operations are described in Sections 9.1 and 9.2.

13.8.1.6 Acceptability for Disposal of TMI Wastes (114)

Low-activity wastes having characteristics similar to those being routinely disposed of by commercial shallow land burial are being disposed of at the Hanford, WA, commercial disposal site. Another commercial disposal site which would be acceptable for disposing of low-activity wastes exists in Beatty, NV. EPICOR-II first-stage wastes are unlike routinely-generated wastes due to high Cs-137 and Sr-90 concentrations, but if they are solidified or placed in a high integrity container, they could be disposed of at an arid site provided emplacement is such that future access to the wastes by inadvertent reclaimers is minimized. However, the most practical alternative is to transfer these wastes to an existing federal government facility for future processing and eventual disposal. High-activity wastes such as the first-stage zeolite liners in the licensee-proposed SDS will be unacceptable for near surface disposal. These high-activity wastes will require on-site or off-site storage until suitable disposal facilities (such as a high-level waste repository) are constructed and licensed. Spent fuel wastes will require disposal in a high-level waste repository. A high-level waste repository is scheduled to be available for disposal between 1997 and 2006. The DOE has the legislative jurisdiction for siting, constructing, and operating high-level waste repositories.

13.8.2 Volume Reduction/Solidification Processes

13.8.2.1 Incineration of Wastes (50, 75, 100, 114)

On-site incineration of wastes is currently only being considered for low-activity (0.006 Ci/ft^3) compactible trash to reduce final volumes by a factor of at least 50. The effluents and releases to the outside environment for incineration of combustible compactible trash are detailed in Section 8.3.

Two types of wastes are generated by incineration; one is the resulting ash generated by burning, and the other is off-gases released from combustion. Resulting ash could be packaged in 55 gallon drums and sent in shielded shipments to a low level burial site. It is estimated that 85% of the non-volatile radionuclides would remain in this ash. Thus, 15% of the total trash non-volatile radionuclide content would be subject to off-gas treatment. Treatment consists of a wet scrubber, followed by a HEPA (high efficiency particulate filter) and a volatile radionuclide absorption system. This cleanup train will result in a decontamination factor of 10^6 to 10^7. Thus, in the most conservative case, the cleanup system will reduce volatile and non-volatile off-gas concentrations by a factor of 1 million prior to their release to the environment. Estimated releases during trash incineration for the four major radionuclides present in combustible trash are given in Section 8.3.

Off-site incineration at a DOE facility is a potential treatment alternative for the high-specific-activity organic resins. Such a facility would be equipped with appropriate effluent controls to reduce releases to levels that are as low as are reasonably achievable and below release standards. The alternative treatment options for the high-specific-activity wastes are discussed in Section 8.1, and specific DOE incineration facilities are identified in Section 9.1.

13.8.2.2 Evaporation/Bituminization System and Liquid Volume in the Condensate

The value of 1.44 is the increase in liquid volume required to be handled in the overhead condensate system for the Evaporation/Bituminization alternative. The value is based on data from BNWL-TR-196 referenced in Appendix G.

13.8.2.3 Vinyl Ester Styrene System Characteristics (50)

It was assumed that the Vinyl Ester Styrene (VES) solidification process would include a dewatering step prior to solidification. This accounts for the increase in volume factor of only 1.5. Dow Chemical Company has tested VES up to 2 x 10^9 rads accumulated dose without degrading the VES polymer. If VES is proposed for use, demonstration of waste form stability under radiation would be required for the specific wastes to be solidified.

13.8.2.4 Volume Increase Factor for Cement Solidification (50)

The volume increase factor for cement was conservatively selected to assure that the solidified resin product would remain a stable solid monolith without crumbing or spalling. It is recognized that a lower volume increase factor could be used provided the licensee demonstrates the proposed formulation will produce a stable monolithic waste form.

13.8.2.5 Solidification and Immobilization of Ion-Exchange Media (20, 50)

Solidification of the ion-exchange media will immobilize radionuclides to a greater extent than will routine packaging of dewatered resins. Brookhaven National Laboratory studies indicate that radionuclides will quickly leach from unsolidified resin beads in ground or salt water leachates. Solidified products, however, would be expected to retain radionuclides better than dewatered resin beads, since the surface area available for leaching has been greatly reduced.

13.8.2.6 Chemical Reactions Between Waste Constituents and Immobilization Agents (78)

The agents and process alternatives discussed for solidifying the waste generated at TMI have been proven through laboratory evaluation and field applications. Chemical compatibility under a variety of conditions has been verified. Additional testing of small batches of TMI wastes, solidified with the selected immobilization agents, will be performed prior to actual large scale processing.

13.8.2.7 Occupational Exposures during Waste Solidification (50)

Occupational exposures, as provided in Chapter 8.0, include the exposure to workers during solidification operations conducted on-site prior to shipment. Where the wastes are shipped off-site for subsequent treatment, the exposures at the off-site treatment facility are not included.

13.8.2.8 Low-Level Waste Act (125)

The Low-level Radioactive Waste Policy Act passed by the Congress in December 1980 and how it could affect TMI is discussed in Section 9.1.

13.8.3 EPICOR-I Wastes

13.8.3.1 EPICOR-I Waste Storage (52)

EPICOR-I liners which were stored in the temporary storage facility were backlogged from the first several weeks of the accident. EPICOR-I wastes have been subsequently removed from storage and shipped. EPICOR-I resins have significantly less activity than do the first stage EPICOR-II wastes.

13.8.3.2 EPICOR-II Waste (76, 100, 121, 123, 130)

EPICOR-II first-stage resin liners contain approximately 30 ft^3 of ion-exchange material and are loaded to a maximum of approximately 1300 Ci. These liners contain bulk specific activities of approximately 45 Ci/ft^3, as opposed to 0.1 to 1 Ci/ft^3 for resins routinely generated by operational reactors. In addition, the EPICOR-II first-stage activity is essentially all Cs-137 and Sr-90, isotopes that have 30-year half-lives. Routinely generated resins, on the other hand, primarily contain Co-60 with a 5.3-year half-life.

The Commission order of October 16, 1979 authorizing EPICOR-II operation states, "The licensee shall not ship spent resins off-site unless they have been solidified, and only then with the prior approval of the Director of NRR (the Director of the Office of Nuclear Reactor Regulation), provided however, that the licensee may ship non-solidified but dewatered spent resins off-site if it determines, and the Director of NRR concurs, that such shipment is required to assure continued operation of EPICOR-II or otherwise required to protect public health and safety." This order is discussed in Section 1.6.1.5.

The NRC has stated that the first stage EPICOR-II resins would be acceptable for disposal at an arid commercial disposal site if they are solidified, or placed in a high integrity container, and if disposal procedures include provisions to minimize inadvertent future contact with the waste. However, the most practical alternative is to transfer these wastes to an existing federal government facility for future processing and eventual disposal.

13.8.3.3 Storage of EPICOR-II Wastes (51, 74, 121)

Depleted resins from the EPICOR-II system are being temporarily stored in a facility referred to as the Interim Radwaste Storage Facility. This facility is a modular structure with each module consisting of approximately 60 storage cells. Each cell can hold one 6 x 6 liner or two 4 x 4 liners. Radiation dose can cause the resins to degrade. However, in the event that corrosion caused the liners to rupture, the contents of the liner would be contained in the storage cells consisting of galvanized corregated steel cylinders with welded steel base plates. The NRC staff, along with the Department of Energy, has anticipated the effects of radiation damage to the resins and is in the process of developing tests to ascertain the condition of the resins currently in storage. The NRC staff has not considered the incineration of the EPICOR-II resins as an alternative process prior to storage because of the high-specific activities which would result from the process.

13.8.3.4 Degradation of EPICOR-II Resins and Container Integrity (20, 53, 59, 75, 76, 74, 84, 93, 64, 100, 114, 115, 120, 121)

Brookhaven National Laboratory (BNL) has indicated that degradation of the ion-exchange material in stored EPICOR-II liners may accelerate liner corrosion and produce a material which agglomerates, making further handling and processing of the material difficult.[1-3] Due to this concern, NRC has contracted with BNL to perform additional tests to evaluate radiation degradation effects

To evaluate the current condition of the stored EPICOR-II resins, the licensee plans to obtain liquid and gas samples from actual EPICOR-II first-stage liners. In addition, the DOE plans to perform detailed tests on EPICOR-II first-stage liners which will include examination of resins for degradation and carbon steel liner for corrosion. The licensee and DOE tests are designed to determine the extent of radiation damage on the ion-exchange material and to compare the actual resin conditions to those projected by testing performed by NRC consultants. The condition of the liner, as determined from these tests, would establish the handling, packaging, and disposal constraints imposed on each shipment. In addition to requiring solidification prior to disposal, the EPICOR-II first-stage resins will require special handling and disposal procedures. The resins are being stored in specially designed concrete storage modules that contain monitored sumps, so that in the event of any radioactivity leakage, mitigating actions could be taken.

13.8.3.5 EPICOR-II Waste Volume Projections (74)

The cumulative number of liners from EPICOR-II operation is projected to be higher than indicated in the October 1979, Environmental Assessment for EPICOR-II System (NUREG-0591). This is in part

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[1] R. Barletta et al., "Status Report on Leachability, Structural Integrity, and Radiation Stability of Organic Ion Exchange Resins Solidified in Cement and Cement with Additives," BNL-NUREG-28286, May 1980;

[2] K. K. S. Pilay, "Radiation Effects on Ion-Exchangers Used in Radioactive Waste Management," NE/RWM-80-3, Pennsylvania State University and Brookhaven National Laboratory, October 1980;

[3] K. Swyler et al., "Review of Recent Studies of the Radiation Induced Behavior of Ion Exchange Media," BNL-NUREG-28682, November 1980.

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[3] K. Swyler et al., "Review of Recent Studies of the Radiation Induced Behavior of Ion Exchange Media," BNL-NUREG-28682, November 1980.

due to the processing of some flushing water used in the decontamination of tanks and equipment in the AFHB. However, the increased number of liners does not have a significantly different environmental impact, in terms of either worker exposures or dose to the public.

13.8.3.6 Special Processing Facilities for EPICOR-II Wastes (76)

The off-site treatment alternatives for EPICOR-II wastes, which include immobilization and elution and reuse for the zeolites and resins and, additionally, incineration and acid digestion for resins, are described in Section 8.1. The facilities at which these treatment options can be conducted, i.e., DOE facilities, are described in Section 9.1. The transportation routes to these potential facilities are provided in Section 9.2.

Off-site storage is a potential alternative that may be considered both in combination with a treatment alternative, or without treatment. The storage alternatives, and constraints on their availability, are described in Sections 9.1 and 9.2.

13.8.3.7 EPICOR-II Resin Packaging and Disposal (64)

The discussion of the alternatives for handling, treatment, storage, and disposal of the wastes generated by the decontamination and cleanup of TMI-2 has been expanded in Sections 8.1, 9.2 and 9.3. In addition, the costs of these waste management alternatives are provided in Section 9.6.

Alternatives of various disposal sites have been examined and are discussed in Section 9.2. These alternatives include use of DOE facilities, certain shutdown commercial sites, or a new regional or intrastate LLW disposal site. It is likely that interim storage of the TMI-2 wastes either on-site or off-site would then be required until an alternative government or commercial site became available.

13.8.3.8 EPICOR-II Resins Relative to 10 CFR 61 Guidelines (74)

The proposed limits for shallow land burial are given in the radionuclide concentration guidelines presented in "Draft Technical Basis for Supporting Additional Technical Criteria and Regulatory Guides to Implement 10 CFR 61 for Land Burial of Low-level Wastes." (This document is a section of the preliminary draft of 10 CFR 61, "Disposal of Low-Level Radioactive Waste and Low-Activity Bulk Solid Waste.")

Wastes containing radionuclides exceeding the guideline values presented in the above document would generally not be acceptable for disposal with a minimum of three feet of cover material, although higher concentrations may be acceptable provided there are sufficient barriers to reclaimer intrusion. Higher concentrations may also be acceptable depending on the form of the wastes, for example, if the waste is solidified.

13.8.4 Waste Storage

13.8.4.1 Storage of Accident Sludges (52, 55)

The treatment, packaging and handling of sludge generated from the cleanup of the Auxiliary and Fuel Handling Building (AFHB) and the Reactor Building (RB) are discussed in Section 8.1. Two alternatives are presented for the packaging of AFHB and RB accident sludge:

- Immobilization of the sludge with cement or vinyl ester styrene in 55-gallon drums yielding a maximum number of packages. Activities would range from 17 Ci/package (RB sludge) to 250 Ci/package (AFHB sludge).

- Dewatering of the sludge and placement in 4' x 4' 45-ft^3-high integrity steel liners yielding the minimum number of packages. Activities would range from 72 Ci/package (RB sludge) to 2,700 Ci/package (AFHB sludge).

Interim storage of dewatered accident sludge liners prior to further off-site treatment and/or final disposal will take place for as short a period of time as possible. Liners will be shielded and handled remotely, and periodic inspection of liner integrity will be performed. Immobilized accident sludge drums will be handled and stored in a similar manner prior to transport to a disposal site.

13.8.4.2 On-site Storage Facilities (31, 70, 71, 75, 125)

The on-site storage facilities (Interim Radwaste Storage Facility and Spent Fuel Pool), are intended for temporary storage of radioactive wastes prior to shipment and disposal off-site. All radioactive wastes, including fuel from the reactor, will not remain or be disposed of on-site over the long-term. It is not intended that, Three Mile Island become a radioactive waste disposal site. The NRC staff has determined that these interim storage facilities are safe alternatives for the purpose of temporary storage.

13.8.4.3 Special Considerations for Temporary On-site Storage (20, 31, 40, 50, 52, 56, 60, 67, 69, 71, 72, 75, 78, 80, 84, 100, 102, 103, 114, 120, 121, 123, 124, 130)

Section 9 includes specific discussions of alternatives for storage of both low-level and high-level wastes. "Storage of wastes" connotes a temporary condition where wastes remain retrievable until final disposition or "disposal" can be performed. "Storage" does not imply the permanent and ultimate disposition of waste.

Temporary storage of wastes at TMI can be acceptable provided that appropriate engineered facilities are constructed. The storage modules for EPICOR-II liners and the spent fuel pool are examples of facilities that have been approved for storage of specific radioactive materials. These modules would be expected to have at least a 20- to 30-year lifetime. Temporary storage of radioactive wastes at a nuclear facility is not a violation of NRC regulations.

Any storage facility for TMI wastes will be designed to minimize environmental effects and will be provided with monitoring systems to detect leaks in waste containers should they occur. Monitoring systems would provide the licensee with early warning of leaking containers so that appropriate measures can be taken to prevent migration of radionuclides.

The geologic and hydrologic characteristics of the TMI site make it unacceptable as a permanent repository for low-level wastes and also for high-level wastes that require isolation for thousands of years. On-site storage of wastes could be required for periods of 20-30 years if adequate off-site facilities are not constructed.

NRC is aware that the designation of disposal sites should be performed as soon as possible. In addition, the shipment and disposal of TMI wastes should also be performed in a timely manner provided that these operations are not inimical to the health and safety of workers and the public. The removal of wastes from TMI depends on the availability of suitable off-site storage facilities or disposal sites. Therefore, a specific cutoff date, after which all wastes would be required to be shipped off-site, cannot be selected.

13.8.4.4 No Regulation to Prohibit Temporary Waste Storage On-site (114)

There is no regulation that specifically prohibits the operation of a nuclear power plant with high-activity radioactive wastes stored outside of the spent fuel pool. However, there are requirements that the storage of these wastes be safe and resulting radiation exposures due to the storage be within regulatory limits and maintained to as low as reasonably achievable levels.

13.8.4.5 Availability of Off-site Storage Facilities for Fuel and TRU Wastes (32, 55, 103)

There are currently transuranic waste storage facilities operated by DOE at the Los Alamos Scientific Laboratory and at the Idaho National Engineering Laboratory. While the DOE has not agreed to use these sites for commercial wastes, the use of these facilities is a technically feasible option. Stipulations imposed by DOE on the licensee would include meeting specific requirements for the packaging of wastes to be stored. Off-site storage alternatives, both for the high-activity solid waste and damaged fuel, are described in Sections 9.1 and 9.2.

13.8.4.6 Spent Fuel Storage Alternatives and U. S. Reprocessing Policy (20, 50, 75, 84)

The damaged and undamaged spent fuel can be placed in storage either on-site or off-site. The spent fuel would be stored until packaging options were developed for disposal in a high-level waste repository or the fuel was reprocessed. It is expected that a portion of the spent fuel will be shipped to various laboratories for research and examination. While reprocessing of spent fuel is a technically feasible alternative, current national policy precludes reprocessing. Should this policy be changed in the future, the reprocessing of the TMI spent fuel would be reconsidered.

The existing spent fuel pool, located in the Auxiliary and Fuel Handling Building (AFHB), could be used for the temporary storage of damaged and undamaged spent fuel removed from the reactor core. Damaged fuel would be stored in special containers to prevent contamination of the water in the spent fuel pool. High activity wastes, such as the zeolite ion-exchange media liners from the zeolite/resin processes used to clean up the liquids in the containment sump, could also be stored in the existing spent fuel pool. It is estimated that space for up to 70 zeolite liners is available in the spent fuel pool. It is not expected that EPICOR-II liners would be stored in the spent fuel pool. Since these liners, which have lower maximum total activities than the zeolite liners, would not require as much shielding. The components of the core suport structure could be cut up into smaller sections, and stored in the fuel transfer canal after transfer of the fuel. In this location, the components should not interfere with other aspects of the cleanup operations.

Dry storage of spent fuel is discussed as a potential storage option in Section 9.1 and 9.2. The use of storage vaults, hot cells, or specially designed spent fuel caissons are possible dry storage options.

13.8.4.7 Storage of Wastes in the Reactor Building (104)

The use of the reactor building for storage of wastes is an alternative discussed in Section 9.2.

13.8.4.9 Storage of EPICOR-II Processed Waste (20)

The EPICOR-II system does not itself remove radioactive waste from the plant site. EPICOR-II decontaminates the water and concentrates the radioactivity in ion-exchange liners.

The EPICOR-II process has proved highly successful in removing radioactivity from the accident water in the AFHB, giving decontamination factors of approximately 10^7 for the two most important fission product elements, cesium and strontium. About 55,000 Ci of radioactivity have been removed from 500,000 gallons of water. The decontaminated water is stored on the site, and the bulk of the radioactivity is contained in spent ion-exchange liners which are stored in the Interim

Radwaste Storage Facility. Disposal of the processed water (containing tritium) and the liners must await final selection and approval of disposal methods. Since the site has not been approved as a waste-storage facility, and probably cannot be so approved, long-term storage of these materials is not contemplated.

13.8.4.10 On-site Storage Facility Shielding Requirements (50)

The on-site interim storage facilities currently in use at TMI, and those under construction, are discussed in Section 9.2. Several facilities will ultimately be used on-site for both storage and staging areas for different types of waste packages. Each would be designed for specific characteristics of the wastes to be stored. Construction of an unshielded interim storage and staging area for the low-level waste packages is contemplated. However, all storage facilities will be required to meet NRC regulatory standards for doses to workers and at the facility boundary, as well as the principle to maintain radiation doses to levels as low as reasonably achievable.

13.8.5 Transportation of Wastes

13.8.5.1 Adherence to Transportation Regulations (25, 75)

Inherent in the approaches to be selected for packaging and transporting the wastes from TMI is adherence to all DOT and NRC regulations that have been developed to minimize the risks associated with transportation of nuclear material. This applies to the design of the selected shipping packages and casks, transportation modes and routes, and operational procedures to be followed under both normal and accident conditions.

Should a vehicle accident occur enroute, the carrier will follow stipulated notification and operational procedures, as defined in an approved emergency plan developed in accordance with NRC and DOT requirements.

13.8.5.2 Transportation Regulations and Procedures (25, 40, 75, 85)

Transportation of radioactive materials is regulated primarily by the Department of Transportation (DOT) and by the Nuclear Regulatory Commission (NRC). The DOT regulations are set forth in Title 49 of the Code of Federal Regulations, primarily in 49 CFR Parts 170-189. The NRC safety regulations are set forth in Title 10 of the Code of Federal Regulations, primarily in 10 CFR Part 71. NRC regulations apply both to persons who ship radioactive materials or who offer them for transport and to carriers who load and transport these materials, providing for protection of both transport workers and transport of hazardous materials, including radioactive materials.

In a recent assessment of regulations (NUREG-0170), the NRC staff found that the risks to the public health and safety from transportation of radioactive materials are very small, concluding that the environmental effects of normal transportation and the attendant risks allow continued shipments by all modes and that no changes to the safety regulations are necessary.

The NRC also has an inspection and enforcement program to assure that licensees do in fact comply with the safety and safeguards requirements during the shipment process. Current inspection plans call for inspection at the point of origin or the point of destination to determine the licensee's compliance with applicable requirements. The NRC is empowered to take enforcement action where licensees are not satisfying NRC requirements.

The NRC requires licensees to give the NRC advance notice for shipments of spent fuel and passes this information on to appropriate state agencies on request.

The NRC has set up a special procedure for waste shipped from the Three Mile Island power station whereby the utility notifies designated persons in each state through which a shipment will pass.

If requested, the utility also informs the state police of pending shipments, giving them vehicle identification, estimated time of arrival, and route data.

As required in Section 301 of Public Law 96-295, the NRC is drafting regulations to require timely notice for governors of states in which will pass shipments of radioactive waste and spent fuel. Such notice will not apply, however, to radioactive waste shipments that the NRC determines do not pose a significant hazard to the health and safety of the public.

The DOT regulations address qualifications of drivers at 49 CFR Part 391. These regulations cover familiarity with rules, prohibition of aiding or abetting violations, driving ability, age, fluency in communicating with the public and understanding highway signs and signals, driving experience or training, driver disqualifications, driver background and character, road tests and written examinations, physical qualifications and examinations, and files and records.

In addition to training required by the DOT, state transportation agencies, and individual employers, the NRC requires for spent fuel shipments that each driver and escort know under what circumstances the cargo vehicle should be immobilized and how to use communication equipment (10 CFR 73.37). In addition, the escort(s) must have completed training in accordance with Appendix D to 10 CFR Part 73. The licensee is responsible for establishing the length and frequency of courses necessary to meet the requirements of Appendix D, 10 CFR Part 73. Section 73.73 of 10 CFR Part 73 does not specifically require fresher courses; however, the licensee is responsible for conducting necessary training to develop and maintain the competency of escorts to accomplish effective emergency response procedures.

13.8.5.3 Proposed Changes in Transportation Regulations (130)

The proposed DOT regulations will not change the special procedure set up by the NRC for advance notice of TMI shipments. Under this procedure, the utility notifies designated persons in each state through which a shipment will pass. The notice may include vehicle identification, estimated time of arrival, and route data, depending on what information each state requests of the utility. The NRC has proposed more general advance notice rules for shipments of radioactive waste materials (45 Fed. Reg. 81058).

The basic effect of the proposed DOT regulations is not to deregulate truck transportation of radioactive materials but to make uniform the routing requirements of such transportation. With these regulations, the DOT intends to reduce the possibility of exposure and inadvertent releases in normal and accident situations in transportation and to clarify the scope of permissible state and local action (45 Fed. Reg. 7140).

The NRC has studied the effect of highway routing controls on accidents in its environmental statement of transportation of radioactive materials (NUREG-0170). By routing trucks on turnpikes or interstate highways, the accident rate would decrease by about ten percent (NUREG-0170, p. 6-12). The number of accidents cited would not change significantly.

13.8.5.4 Possible Illegal Waste Shipments (64)

The wastes generated at TMI-2 are carefully accounted for on-site, and manifested for off-site shipment. The NRC, utility, and carrier maintain detailed records for each shipment, as does the disposal site. Records are available for tracking the waste from its generation through disposal. Illegal shipments from TMI-2 are extremely unlikely, and none have ever been detected.

13.8.5.5 Commercial Shipment of Wastes (130)

Shipment of wastes to the commercial disposal sites will be performed by commercial shippers in accordance with DOT and NRC regulations. Some shipments of radioactive material to DOE research facilities may be performed for the government by commercial shippers or in special cases in government-owned and operated vehicles and equipment.

13.8.5.6 Shipment of Small Quantities of Contaminated Liquids (50)

Shipments of small test quantities of contaminated liquids and ion-exchange materials for laboratory examination, packaged and handled in accordance with NRC and DOT regulations, are permissible.

13.8.5.7 Risk Associated with Cross-Country Shipment of Wastes (45)

The risk in transporting waste, is linearly proportional with distance traveled. Although the risk of transport for a short trip would be less than that for a longer trip, however, in either case, the risk would be small.

The current "most likely" truck route from TMI to Hanford, Washington, is 2750 miles long. An evaluation has been made of the exposures that individuals along the transportation route receive from the waste shipments from TMI. The results of this evaluation, which are provided in Section 9.5.1.2, show that individuals living along the route are being exposed to very small (< 0.01%) increases in radiation levels over natural background for the maximum potential number of waste shipments.

13.8.5.8 Modes of Waste Transport (43)

Truck shipment is the current transportation mode for the TMI-2 wastes, and it is anticipated that it will be used for future waste and fuel shipments. Rail and intermodal transportation are mentioned in the context of available alternatives but, because of their relative constraints and disadvantages, (Section 9.2) they are not presently being considered for shipment of any waste or fuel. Should this situation change, the environmental effects of these modes would be evaluated separately.

13.8.5.9 Transportation Routes (3, 9, 32, 50, 51, 70, 86)

Only one shipment of radioactive solid wastes has traveled via either Route 11 or Route 15. These two roads will not be used for future shipments.

Section 9.2 provides a discussion of transportation alternatives and routing which includes the potential routes to all viable off-site storage, treatment, and disposal facilities, not only to the Hanford facility.

The routes for transporting the radioactive waste to an off-site waste storage, treatment, or disposal facility are selected on the basis of current DOT and NRC guidelines. The interstate highways, in conjunction with urban bypasses, represent "preferred" routes. The transport routes are selected after considerable study, and in consulation with state and municipal officials. Prior to individual shipments, when shipping plans are confirmed, factors such as temporary hazards enroute are considered and, if necessary, modifications are made in the plan to avoid them.

Section 9.2 shows the "most likely" transportation routes to all the possible waste treatment, storage, and disposal facilities. Since a number of interstate, U. S., and state highways are potential routes, it would not be appropriate to single out one route, such as I-80, for a separate accident study. Accident statistics are a factor in selection of all the routes.

The selected routes are subject to review by state agencies and to revision to assure that impacts from normal transportation and potential accidents are minimized. Therefore, the selected transportation routes will incorporate any requirements imposed by the governmental bodies of the state being traversed. The routes to all possible waste treatments, storage, and disposal facilities are presented in Figure 9.2-9.13.

13.8.5.10 Radiation Exposure during Waste Transport (25, 60, 66, 70, 72, 75, 98, 107, 125)

The occupational exposures calculated for the waste truck drivers are presented in Section 9.5. The analysis is based on two drivers making 30 trips per year on each of two potential routes--the longest route of 2,750 miles to Hanford, Washington; and the shortest of 370 miles to West Valley, New York. The annual crew dose is 11.0 person-rem for shipments on the longest route, and 1.6 person-rem for shipments on the shortest route. The occupational exposures truck drivers receive are regulated, and each individual is monitored to assure that the established limits are not exceeded.

In determining the 1.3-mrem exposure of members of the public standing three feet from a stopped loaded truck, a three-minute exposure duration was assumed. During the course of a shipment, it has been conservatively assumed that ten people would be so exposed, resulting in a total population dose of 0.013 person-rem.

As an upper bound the transportation doses to people residing along the shipping route from TMI to the Hanford disposal site for the minimum and maximum number of shipments have been evaluated to range from 16 person-rem to 50 per-rem (Section 9.5.1.2). The approach used for determining such exposures is based on the analysis provided in NUREG-0170, "Final Environmental Statement on the Transportation of Radioactive Material by Air and Other Modes."

Occupational exposures have been determined for all the workers involved in the waste management steps, including those handling and loading the waste packages onto the trucks and the health physicists monitoring these activities. These exposures are provided in Section 9.5.

The exposure of transport workers during loading and shipment is not comparable to exposure of individuals in the general public. Occupational radiation exposure limits are higher because of several reasons; for example, radiation workers are closely monitored for radiation exposures, records of cumulative radiation doses are maintained, and the work they perform are under the provisions of the licensee's radiation protection plan procedures.

The dose to people living near the disposal sites has not been calculated for specific TMI waste shipments since the TMI waste does not add to the waste disposed of at the facility, but replaces other waste shipments. Thus, the TMI waste does not add to the exposure to be received by the population in the vicinity of those facilities.

The potential for transportation accidents and the impacts associated with both airborne and waterborne releases from such accidents have also been evaluated (Sect. 9.5). Release fractions of radionuclides are calculated based on conservative assumptions of severe accident conditions, and a representative inhalation dose determined for this type of accident. Under this "worst-case accident scenario, whose potential frequency of occurrence is in the order 10^{-9} accidents per truck mile for the longest route (2,750 miles), the inhalation dose is less than 10% of the annual dose due to background radiation. Thus, in the unlikely case an accident does occur, the effect of radiation release is not considered to be a significant impact.

The statistics for accident rates used in Section 9.5 were obtained from accident rate statistics in "Transportation of Radioactive Material by Air and Other Modes," NUREG-0170.

13.8.5.11 Responses to Transporation Accidents (25)

Currently, if an accident occurs, state and local governments are primarily responsible for overseeing the response of carrier, shipper, and others and for taking any actions deemed necessary to protect health and safety. To assist state and local governments, the federal government has a program called the Federal Radiological Monitoring and Assessment Plan (formerly called the Interagency Radiological Assistance Plan) which could be used to assist state and local authorities. The program is coordinated by the Department of Energy (DOE). The DOE charges eight regional coordinating offices with the responsibility and authority to convene radiological assistance teams. When called, a team reports to the scene of an accident or other radiological emergency and assists the emergency response personnel already on the scene.

The subject of emergency preparedness in transportation of radioactive materials is under active consideration by both the NRC and the DOT. A joint NRC/DOT study group (NUREG-0535) recommends several actions, including federal rulemaking, on response planning by shippers, carriers, and state and local agencies. The group recommends that state and local agencies develop plans to both advise and assist the carrier and to take appropriate control actions at the scene of an accident to protect public health and safety.

To assist state and local governments in planning emergency responses to radiological incidents at fixed sites or in transportation, the federal government has an interagency program to coordinate planning, guidance, and training (44 Fed. Reg. 69904). In this program, the DOT supplies guidance on emergency response planning related to transportation of radioactive materials. The Federal Emergency Management Agency (FEMA) is the lead agency in this program.

13.8.5.12 Transportation Accident Parameters (120)

An analogy drawn between the behavior of solidified TMI waste and the waste immobilized in glass in Section 9.5 indicates that the fractional release and dispersion of airborne particles would be similar to those under the conditions of a severe truck accident causing container rupture and attendant fire explosion.

Two accident cases described in Section 9.5, the release of airborne respirable particles from a ruptured container with a fire and the release of waterborne material from a ruptured container underwater, are considered to be boundary cases for maximum impact of potential accidents. The release fractions used have been analyzed using highly conservative assumptions, since Type B transportation packages, which would be used for the high-activity-wastes, are designed to withstand the effects of a 30-minute diesel-fuel fire without damaging the contents. A water-immersion test is also included for this package.

13.8.5.13 Evaluation of Transportation Accidents (66, 25, 107, 125)

Transportation accidents are addressed in Section 9.5.

The environment of the core during the TMI-2 accident was significantly different from the environment of the fuel in a transportation accident would be. The TMI-2 core was apparently uncovered for over half its length for a time long enough to allow extensive damage from overheating. The high temperatures that occurred caused fuel clad failure and the high fractional release of the more volatile fission products. On the other hand, in a transportation accident, even with loss of coolant for spent fuel, both temperatures and the release of fission products would be much lower.

13.8.5.14 Accidental Releases from a Breached Cask (75)

Although the Type B casks are subjected to a test program inclusive of a 30' drop test, they are designed to withstand stresses associated with greater drop distances. In the highly unlikely event that a cask is breached as a result of the postulated 60' drop from within the building the building effluent control systems are designed to handle the airborne effluents, maintaining outside releases within regulatory limits.

13.8.5.15 Shipping Radioactive Wastes in Lower Concentrations (47)

Shipping radioactive wastes in less concentrated forms would not necessarily lessen the total potential environmental impact. Rather, an increased number of shipments, increased radioactive waste handling, and a potential increase in total worker exposures would result. An increased number of shipments may also result in a higher potential for accidents and subsequent releases of radioactivity to the environment. The current practice of radioactive waste shipments with adequate safeguard features (i.e., waste immobilization, radiation shielding and container integrity requirements) has a smaller potential for environmental impact.

13.8.5.16 Vehicle Driver Exposure (130)

Routine exposure to both truckdrivers and members of the general public from transportation of radioactive waste materials from TMI are discussed in Section 9.5. A truck crew member would receive about 60 mrem per trip for the TMI wastes for the longest shipping route. Since the TMI wastes are to be shipped under exclusive-use arrangements (a regulatory term for shipment with only one consignor and only one consignee), the radiation level inside the cab is limited by both NRC and DOT regulations to 2 mrem per hour. Assuming a crew made round-trip shipments to Hanford, WA, every ten days, the maximum number of trips which could be made in a year is 36.5. At 60 mrem per trip, the dose to each crew member would be 2.2 rem. This value is less than the occupational exposure limit in 10 CFR Part 20.

13.8.5.17 Availability of Shielded Shipping Casks (75)

Based on the current cleanup, decontamination, and waste disposal schedules and estimates to packaging waste generation, it does not appear that the availability of casks will pose severe constraint on the waste shipment schedule. However, as noted in Section 9.1 the purchase or lease of additional shielded casks is being considered to alleviate any possible constraints on the schedule.

13.8.6 Waste Disposal

13.8.6.1 DOE to Engage in Radwaste Disposal Evaluations (78)

The question of where commercial radioactive wastes, including those from TMI, will be sent for long term storage/disposal has been studied by both the Department of Energy (DOE) and the NRC. The general conclusions of these studies are that radioactive waste can be safely disposed of, that present plans for establishing disposal sites are established. The details of these studies can be found in several documents, among which are the "Statement of Position of the United States Department of Energy" in regard to "Proposed Rulemaking on the Storage and Disposal of Nuclear Waste" dated April 15, 1980 (DOE/NE-0700) and the Final Environmental Impact Statement on "Management of Commercially Generated Radioactive Waste (DOE/EIS-0466F, 3 volumes).

DOE is also actively engaged in the evaluation of several aspects of the alternatives for the ultimate disposal of the radioactive wastes and nuclear fuel material. This includes the testing of waste forms, selection of suitable sites, and methods of disposal of both the solid radioactive wastes and the fuel from the reactor.

13.8.6.2 Determination of Disposal Impacts (24, 64)

The specific disposal methods and sites for TMI-generated wastes will be selected based on the characteristics of the individual waste materials. Even though not all the specific disposal methods and sites have been selected for TMI wastes, it is possible to identify alternatives and establish bounding estimates for overall differential disposal costs and potential health effects to workers and the public. Inflation costs have not been considered by escalating base year costs using an estimated inflation factor.

13.8.6.3 Waste Disposal Should Not Be a Prerequisite to Cleanup (50, 98)

Although questions regarding offsite disposal still exist, especially with regard to high-specific-activity wastes and spent fuel. However, the NRC staff has indicated that resolution of off-site disposal questions should not be a prerequisite to proceeding with on-site cleanup activities.

13.8.6.4 Low-Level Waste Disposal Site Availability (11, 16, 32, 50, 51, 53, 55, 61, 64, 67, 70, 75, 79, 84, 85, 92, 93, 98, 101, 107, 114, 115, 123)

The commercial disposal site at Hanford, WA, is currently accepting certain wastes generated at TMI. The wastes which have been shipped to Hanford include contaminated trash, some solidified decontamination solutions, and EPICOR-I resins. In November 1980, an initiative was passed in the State of Washington which would prohibit the Hanford commercial disposal site from accepting out-of-state non-medical wastes after July 1, 1981. Agreements between other states and the State of Washington, however, could be formulated which would allow the acceptance of wastes from those other states.

In the event that existing commercial low-level waste disposal sites are unavailable for TMI wastes, storage of these wastes either on-site or off-site would be required until an alternative disposal site becomes available. On-site or off-site storage would be required for wastes requiring disposal in a high-level waste repository until such a facility becomes available. It is intended that the cleanup of TMI would proceed and not be predicated by selection of ultimate disposal methods.

13.8.6.5 West Valley (125)

The West Valley disposal site is considered as an alternative for disposing of TMI generated low-level wastes. However, there are several institutional and technical issues which must be resolved prior to reopening West Valley for TMI wastes.

13.8.6.6 Quantities of Waste Generated (32, 50, 75)

The volume of each waste form generated from various sources during the cleanup and decontamination operations was determined from an evaluation of the treatment or conditioning process characteristics, in conjunction with the quantities of input material to be processed. Experience at other facilities undergoing decontamination, results of development testing at industrial and government facilities, and actual processing experience have provided the process parameters used in each case. The waste volume generated are tabulated for each waste form in Section 8.1.

13.8.6.7 Reactor Systems Wastes (50)

The evaluation of waste types and volumes generated in the reactor system cleanup or core removal considers that all of the system components, both major parts and miscellaneous hardware, that will be disposed of as waste. The type and quantities of waste are described in Section 8.1.

13.8.7 High Level Waste

13.8.7.1 Final Disposition of High-Specific-Activity Wastes (52, 64, 75, 76, 85)

Section 9 includes specific discussions regarding the alternatives for the disposition of the SDS wastes. NRC agrees that the first stage zeolite wastes, which may have Cs-137 at specific activities on the order of 1000 Ci/ft, are unacceptable for routing shallow land burial methods.

Wastes which are unacceptable at the commercial low-level waste disposal sites due to very high specific activities would need to be disposed of using methods which provide greater isolation from the environment than does commercial land burial. The only option which is currently being developed and that would provide the required degree of isolation, is the high-level waste repository. Therefore, high-specific-activity wastes are being considered as candidates for the high-level waste repository.

13.8.7.2 Disposal of High-Level Wastes and Spent Fuel (11, 13, 16, 20, 51, 52, 60, 64, 67, 71, 76, 79, 85, 92, 93, 100, 107, 109, 114, 115)

The DOE has the legislative responsibility for siting, building and operating high-level waste respositories. The DOE currently is performing field studies in several locations throughout the U. S. to determine potential sites for a high-level waste repository e.g., salt domes in Louisiana and Mississippi, in basalt formations in Hanford, Washington, in bedded salt in New Mexico, and tuff formations at the Nevada test site. Based on current plans, it is expected that Final Environmental Impact Statements will be completed for a potential repository for salt domes in July 1983, basalt formations by February 1983, bedded salt in September 1984, and for the Nevada test site tuff formations in November 1984. Following a review of the characteristics of the potential sites in various media, a site will be selected for the first repository. The current schedules include availability of the first repository for commercial high-level wastes between 1997 and 2006.

On-site or off-site storage would be required for wastes requiring disposal in a high-level waste repository until such a facility becomes available. Storage facilities for high-activity wastes would be designed to minimize environmental impacts over a 20- to 30-year storage period to account for possible delays in repository operation.

Currently, spent fuel from nuclear power plants is stored in fuel pools similar to the TMI fuel pool. This spent fuel also requires continued storage until a high-level repository is sited, constructed, and licensed. Therefore, the spent fuel at TMI will be disposed of in a similar manner to the spent fuel which exists at other nuclear power plants. The damaged spent fuel will, however, require special packaging for transportation and possibly special processing to meet the waste form requirements for the high-level waste repository.

13.8.7.3 High-Level Waste Repository Site Selection (51)

The distinction between site selection for a high-level waste disposal facility and site selection for a nuclear power plant, is found in the requirements for storage of the waste on or near the surface in a retrievable manner versus the ultimate deep geologic disposal in a high-level waste repository. Deep geologic disposal requires that the site, located thousands of feet below the surface, isolate the waste from the environment for thousands of years. The basic site requirements for a geologic repository are geologic stability, limited flow of ground water and isolation from the activities of future generations of human beings.

13.8.7.4 DOE Acceptance of High-Activity Wastes (48, 52, 59, 80, 105)

DOE has the facilities, capabilities, and expertise in the processing, storage, and disposal of the high specific activity wastes. The issue of DOE acceptance of these wastes involves the combining of defense and commercial wastes and the possible NRC licensing of these activities.

In this regard NRC Chairman Ahearne, in a letter to DOE, has written to Secretary Duncan on October 20, 1980 and has requested DOE assistance for the processing, storage, and disposal of some of the high-activity waste forms which will be generated in the TMI cleanup. In a follow-up letter, the NRC Executive Director for Operations wrote to the Under-Secretary of DOE and indicated that NRC licensing jurisdiction would not include DOE facilities whose primary function remained the handling of defense and research waste materials.

13.8.7.5 Future Status of DOE (102)

The DOE facilities which are identified as options for waste processing and disposal are involved in national defense activities. These facilities would continue to be operated whether under the jurisdiction of the DOE or under another government agency and TMI would not become a radioactive waste disposal site.

13.8.7.6 Submerged Demineralizer System (SDS) Waste Disposal (70)

The licensee has proposed to use the submerged demineralizer system (SDS) for processing the liquids in the reactor building sump. (The NRC has not authorized the use of the SDS.) The SDS uses different components than the EPICOR-II. The SDS and EPICOR-II, however, utilize ion-exchange as the mechanism for cleanup of the radionuclides in the liquids.

The SDS first stage will generate an inorganic ion-exchange material (zeolite) having a specific activity much higher than wastes which are routinely generated at other nuclear power plants. Because these wastes will be unacceptable for disposal at commercial shallow land burial sites, storage on-site will be required until either the DOE accepts these wastes for processing with their own similar wastes or until acceptable disposal options, such as a high-level waste repository, is licensed and becomes operational.

13.8.7.7 High Specific Activity Waste (53, 76, 98)

For the ion-exchange wastes whose specific activity and inventory of radionuclide contamination requires that they be handled as High Specific Activity Waste (HSAW) e.g., the first stage of the SDS system, off-site storage, treatment, and disposal options will be different from those for low-level waste (LLW).

The off-site treatment alternatives for this waste, which include immobilization and elution from the zeolites and resins and, additionally, incineration and acid digestion for the resins, are described in Section 8.1. The DOE facilities at which these treatment options can be conducted, are described in Section 9.1. The transportation routes to these facilities are provided in Section 9.2.

Interim storage of HSAW and fuel, on-site and probably off-site, are viable and necessary options. The cleanup of TMI-2 is not considered to be dependent on the selection of an ultimate disposal site (i.e., geologic repository) for this material. The cleanup can proceed and, as required, the various waste forms can be stored until the selected treatment/disposal option is available. Off-site storage may be considered for HSAW both in combination with a treatment alternative, or without treatment. The storage alternatives, and constraints on their availability, are described in Sections 9.1 and 9.2. The routes to these facilities are provided in Section 9.2.

The alternatives and existing constraints for disposal of HSAW are discussed in Sections 9.1 and 9.2. Since the prime option for disposal of the HSAW is in a geologic repository, specific locations cannot be identified pending selection of potential repositories.

13.8.7.8 Vitrification of Ion-Exchange Media (50, 55, 76, 121)

The discussion of a treatment alternative for the high-specific-activity ion-exchange material is found in Section 8.1. Among the treatment alternatives discussed is immobilization by vitrification in glass, which would be conducted at off-site facilities designed to process high-level waste streams. Specific facilities at which these immobilization capabilities exist are identified in Section 9.1. Facilities available for immobilizing the HSAW in a glass or ceramic matrix are located at DOE sites.

Representative costs for each of the viable treatment, packaging, and disposal alternatives are discussed in Section 9.6 and Appendix G. Thus, it will be possible to weigh costs along with technical feasibility and environmental impacts to evaluate an alternative.

13.8.7.9 Acceptability of High-Level Wastes at Hanford (92)

High-level wastes would be unacceptable for disposal at the Hanford commercial disposal site, notwithstanding the initiative passed by the Washington State Electorate in November 1980. High-level wastes would require disposal in a high-level waste repository which is to be developed by the DOE. DOE is currently investigating suitable geologic media at several sites for locating a high-level waste repository. The first repository is scheduled to be available for disposing of wastes between 1997 and 2006.

13.8.7.10 Special Nuclear Material In Wastes (50, 100)

Special nuclear material (SNM) is defined in 10 CFR 70 as: "(1) plutonium, uranium-233, uranium enriched in the isotope 233 or in the isotope 235, and many other materials which the Commission, pursuant to the provisions of Section 51 of the [Atomic Energy Act], determines to be special nuclear material, but does not include source material; or (2) any material artificially enriched by any of the foregoing but does not include source material." The regulations in 10 CFR 70 do not provide numerical concentration or mass limitation exemptions for SNM.

The commercial disposal sites are currently licensed to accept for disposal waste materials containing less than 10 nCi/gm of TRU materials including plutonium. Wastes containing greater than 10 nCi/gm would require storage until either a high-level commercial waste repository or a TRU disposal facility becomes available.

Disposal site license conditions also include specific possession limits for uranium-233. These possession limits vary at each disposal site and are based on criticality considerations. Since most uranium-233 is derived from the thorium fuel cycle, which is not applicable to TMI, uranium-233 will not limit disposal considerations.

The source of plutonium in the wastes would be fuel debris, which could be removed as particulates in the processing of the reactor building water or the primary system liquids.

13.8.8 Other Comments

13.8.8.1 Comparison of Requirements for Processing and Disposal of Wastes at TMI and at Other Nuclear Power Facilities (130)

Those TMI wastes which are similar to routinely generated wastes are acceptable for disposal at commercial shallow land burial sites. These wastes include contaminated trash, some solidified decontamination solutions, and some of the EPICOR-II resins.

EPICOR-II first stage resins, however, contain bulk activities of approximately 40 Ci/ft^3 as opposed to 0.1 to 1 Ci/ft^3 for routinely generated resins. Also, the EPICOR-II first-stage activity is essentially all Cs-137 and Sr-90, isotopes that have 30-year half-lives. Routinely generated resins, on the other hand, primarily contain Co-60 with a 5.3-year half-life. Because of the higher concentrations and longer half-lives, EPICOR-II first-stage resins would require solidification, or placement in a high integrity container, prior to disposal as well as special handling operations and disposal procedures at commercial disposal sites. The most practicable alternative is transfer these wastes to an existing federal government facility for future processing and eventual disposal. First-stage ion-exchange material from the licensee proposed system to process the reactor building sump liquids will have Cs-137 concentrations of approximately 1000 Ci/ft^3. Wastes having these characteristics will be unacceptable for routine shallow land burial.

13.8.8.2 Foreknowledge of the Characteristics of Unique Accident Waste (32)

The staff has reviewed material, including documents and testimony of individuals, as well as pro- and anti-nuclear groups pertaining to the probabilities of severe accidents at nuclear power facilities. None of this material provides specific characteristics of waste products that might be generated during the post-accident cleanup to any great detail useful for the cleanup.

13.8.8.3 Possible Use of Salem Nuclear Power Facility as a Disposal Site (48)

Radioactive solid wastes from the evaporation and the zeolite/resin alternatives for processing liquids at TMI will have to be disposed of at either licensed commercial or government disposal facilities, which have acceptable geologic hydrologic characteristics for the disposal method used. Currently, the only licensed commercial disposal facilities are located in Barnwell, SC, Beatty, NV, and Hanford, WA. The geologic/hydrologic characteristics of the Salem site (shallow ground water table and oceanside location) would likely prevent it from being acceptable as a disposal site for TMI wastes.

13.8.8.4 Access to Proprietary Data (53)

Under NRC regulations, NRC can obtain and use proprietary data affecting safety reviews and provide protection of a firm's proprietary interests. In most cases, commercial firms freely release the information requested by NRC. For firms that do not release proprietary data freely, NRC has subpoena power to obtain the information necessary to prepare safety reviews.

13.8.8.5 Decontamination and Disposal of Reactor Coolant Pumps and Motors (75)

The hazards of handling, transporting, and disposing of all wastes, including the reactor coolant pumps and motors, will be considered to assure compliance with NRC and DOT regulations.

13.8.8.6 Use of Metal LSA Boxes (50)

Metal LSA boxes are a viable alternative to the use of wooden boxes, cost and availability being considerations.

13.8.8.7 Clarifications (75)

The term "remedial activities" as used in the PEIS intended to apply to the various cleanup activities (other than decontamination) currently in progress at TMI.

The term "current phase of operations" used in the PEIS relates to the work performed to date (January 1981) at TMI-2 in decontamination and other cleanup operations as far as the management of the generated waste is concerned.

13.9 RADIOLOGICAL EFFECTS

13.9.1 Releases & Effluents

13.9.1.1 Prudent vs Zero-release Levels (125)

A zero-release objective is unattainable and does not provide useful guidance for establishing guidelines or choosing between alternatives. It is more appropriate to choose those alternatives that will minimize all of the environmental impacts (not just those that result from releases) and hazards for all of the cleanup operations over the entire cleanup period and beyond. The best method is to: (1) analyze and divide the overall cleanup project into a series of operations and tasks that are not too strongly interdependent, (2) identify reasonable alternative approaches for each, and (3) choose those alternatives for which the environmental impacts and hazards are "as low as reasonably achievable" (ALARA) as the cleanup proceeds. In many cases the data needed for an informed judgement for a particular cleanup task will not be available until prior cleanup tasks have been completed. However, the complex situation and the limited amount of information available require an approach in which the experienced personel responsible for the cleanup operations make many specific decisions. The NRC will, of course, retain the right to review and approve specific plans as they are developed.

If the dose any individual receives from a particular release were very small compared to the doses for natural sources of radiation, it would not be reasonable to undertake a major expenditure to further reduce the release. The "ALARA" concept used in the PEIS recognizes and deals with such situations; the "zero-release" concept does not.

13.9.1.2 Present and Future Abundances and Locations of Radionuclides (64, 74, 75, 79, 84, 85)

The total quantities of fission products and actinides in the TMI-2 reactor at the time of the accident, and for any subsequent time thereafter, are accurately known from detailed calculations of the buildup and decay history of the individual isotopes. Tritium, cesium and strontium are present in 3 places: namely 1) the primary cooling system, 2) the Reactor Containment Building (RCB) Sump and 3) the EPICOR II resins (which removed the radionuclides from the AFHB water). In addition, there is still some tritium left in the processed AFHB water. Radionuclides are present in one of the following four forms: 1) a solute in water; 2) "plateout" on surfaces of the cooling system or RCB sump; 3) sludge in the AFHB and the RCB sump and 4) suspension of fine particles (about 5% of the sump concentrations).

In chapters 7 and 8 a detailed mass balance is given showing the source and final disposition (waste form) of the radioisotopes in the RCS and the sump water, and also those radioisotopes which entered the RCS during defueling. The quantity of sludge, filterable solids and "plateout", and the way in which it is treated are discussed in Sections 5 and 8.

13.9.1.3 Best and Worst Cases of Processed Water Activity (13)

Estimates of the concentrations of the radionuclides in the processed water are dependent on the water treatment system (Section 7.3.1.2).

The best and worst case treatment systems were identified and used to characterize processed Reactor Building and Reactor Coolant System water. The best case, which led to the effluent with the lowest concentrations of contaminants, was the SDS/EPICOR II treatment system. The worst case, which led to the effluent with the highest concentrations of contaminants, was the SDS treatment system. These systems and other alternatives are discussed in Section 7 and Appendix G.

13.9.1.4 Results of Reactor Building Purge Samples (73)

Airborne particulate samples, released during the purge of the TMI-2 reactor building atmosphere, were analyzed for gross beta activity by EPA and the licensee. All results were less than the minimum detectable levels (0.1 pCi/m^3 of air sampled for the EPA). NRC also estimated the Sr-90 and Cs-137 airborne particulate radioactivities to be less than 0.06 μCi each in the TMI-2 reactor building prior to purging. Following the purge, EPA performed radiochemical analysis for the combined filters for strontium-90, but those results are not yet available.

13.9.1.5 Discharge of Krypton (20)

Information regarding the purge of krypton from the TMI-2 reactor building atmosphere is available in a separate comprehensive environmental review NUREG-0664.

13.9.1.6 Alternatives for Disposing of Processed Water (85)

Section 7 provides a detailed assessment of the alternatives for the controlled release of the processed water, including facility, effluent and radiological doses under normal operation and accident conditions. Other potential impacts such as psychological stress and socioeconomic impacts are also addressed.

13.9.1.7 Radioactivity Levels for Processed Water Release Alternative (12, 72, 79)

At present, the NRC has not authorized any alternative for the release of processed accident contaminated water from TMI-2, including the alternative for the controlled release of processed accident water into the Susquehanna River. Should the NRC decide to authorize such release, it will ensure that the licensee implement the release in a controlled manner, according to the conditions (technical specifications) and criteria determined by the NRC, to protect the public health and safety and the environment. In particular, the concentration of radioactivity in the water at the nearest drinking water intake downstream of TMI would be diluted to levels below EPA Drinking Water Standards. Also, the absorbed radiation dose in fish and seafood in the river and Chesapeake Bay area would be an insignificant fraction of the normally occurring natural background radiation dose and should cause no detectable biological effect. (Section 7.2.)

13.9.1.8 Additional Processing of Decontaminated Water (52)

Processing of radioactive waste liquids would be required to follow the principle of maintaining radioactive material releases to as low as reasonably achievable (ALARA) levels. Thus, when the processing alternative meets the numerical design objectives of Appendix I to 10 CFR 50, as proposed by the Staff in Section 1.6.3.2 and Appendix R, further processing of the processed water would not be considered to be necessary. One alternative discussed for the disposal of processed water is by forced evaporation via a heated pond. A suggestion to consider such an evaporation "in a container, with filtered vapor venting, (that) would allow release of essentially nothing but tritium, and would allow non-volatiles in the bottom water to be solidified" is analogous to considering an additional processing step for the processed water which already meets the ALARA principle. Such additional processing is not considered to be necessary and has not been included in the discussion.

13.9.1.9 Factors for River Dilution Considerations (52)

For releases of long duration, it is possible to relate the steady state dilution of Susquehanna River water into the Chesapeake Bay to the observed salinity. See Section 7.2.5.4 for a more detailed discussion.

13.9.1.10 Interaction of Tritium and Other Radionuclides with Sediment (46)

The behavior of tritium in the form of tritiated water (HTO) is practically identical to that of ordinary water (H_2O). Tritium does not settle and remain in sediment any longer than does the hydrogen in ordinary water. Of the environmentally significant radionuclides which could be released from TMI, only Cs-134 and Cs-137 interact strongly with sediments. In any event, the radiocesium would be present in much lower concentrations in the sediments of the Chesapeake Bay than deposits due to other sources such as nuclear fallout or routine releases from operating nuclear plants. The behavior of these nuclides is discussed in Section 7 and Appendix T.

13.9.1.11 Distribution of Radionuclides in River (13)

The calculations of radionuclides in the Susquehanna River and Chesapeake Bay have been revised. Calculations of radionuclide concentrations in fish flesh near TMI consider the limited flow of the center channel of the Susquehanna River. Radionuclides will be completely mixed at the points in the river where drinking water supplies are taken, for the reasons given in Section 3. An analysis of radionuclide concentration in fish near TMI appears in Section 7.2.5.4.

13.9.1.12 Effect of Sediment in Radionuclide Distribution (21, 64, 74, 75, 79, 85)

The importance of sediment on the radioecology of the Susquehanna River and Chesapeak Bay is discussed in Section 7 and Appendix T. Of those radionuclides interacting with sediment, the most radiologically important are Cs-134 and Cs-137. A conservative calculation of the effects of sediment on the transport of radiocesium released from the Peach Bottom Nuclear Plant was performed in Appendix T. This calculation is based on measurements of Cs-134 and Cs-137 in sediment and fish of Conowingo Pond and the upper Chesapeake Bay. The staff estimates that no more than 12 percent of the radiocesium released from the Peach Bottom Plant became associated with the sediment. The effects of sediment on the radioecology of the Susquehanna River and Chesapeake Bay will therefore be minor. Gross (1978) estimates that sediment is eroded from the Susquehanna River for flows of 400,000 cfs or greater. Flows of this magnitude would have a recurrence interval of about six years. Dilution of resuspended sediment would be great for such large flows.

13.9.1.13 Location in River/Bay Where Radioactivity Can Be Detected (42, 52)

Except in the immediate vicinity of TMI-2, it is unlikely that any radioactivity released from TMI will be detectable in the Susquehanna River or in the Chesapeake Bay above naturally occurring levels caused by nuclear fallout or normal releases from other nuclear plants. See Appendix T.

13.9.1.14 Release Estimates from Trash Incineration Alternative (83)

Release estimates from trash incineration are provided in Section 8.

13.9.1.15 Source of Tritium in Sample Wells (69)

Tritium has been detected in sample wells at the TMI-2 facility. The licensee and NRC believe, from the isotopic analysis results and from the location of the samples, that the leaks were from the valves of the outdoor Borated Water Storage Tank, and not the reactor building. The licensee has a continuing program to monitor and rectify the source of leaks.

13.9.1.16 Radioactivity Releases from TMI Prior to the Accident (107)

The radioactivity effluents from TMI were monitored and reported to the NRC in compliance with the requirements of the TMI technical specifications. The offsite radiation doses during the period prior to the accident were within the regulatory limits of 10 CFR Part 20 and met the numerical design objectives of Appendix I to 10 CFR Part 50.

13.9.1.17 Unplanned Releases Prior to the TMI-2 Accident (99)

There were no unplanned releases reported to the NRC by the TMI-2 facility prior to the March 28, 1979, accident.

13.9.1.18 Radioactivity Releases Since the Accident (60, 100)

Since the accident, the only significant release of radioactivity was the controlled purging of Kr-85 from the reactor building. The total amount of Kr-85 released was about 43,000 Ci. "Miniventings" may be required as long as radioactive waste water remains in the reactor building unprocessed. This requirement is adressed in Sections 4 and 6.

13.9.1.19 Releases of Radioactivity to River from Other Facilities (107, 112, 20)

The EPA Drinking Water Standards apply at the point of drinking water distribution regardless of the number of sources of discharge. Therefore, safety of the drinking water is assured. In addition, the NRC subscribes to the EPA 40 CFR 190 requirement on radiation resulting from the nuclear fuel cycle, which includes the radiation dose received from all pathways, i.e., airborne, liquid, and direct radiation, and which includes cumulative radioactive releases.

13.9.1.20 Precedents for Radioactive Liquid Releases from Operating Reactors (115)

Precedents have been established for the release of radioactive water into rivers. Normal operating effluents are currently being released into the Susquehanna River and other bodies of water. For example, the average amount of tritium released from a normal generating unit of TMI-2 size is 400-500 curies/year. These planned releases are limited by the conditions of the operating license and must be reported to the NRC on a periodic basis.

The guidelines on these releases are such that the effects on fish, wildlife and, ultimately, man are in compliance with the NRC design objective of limiting exposure to "as low as is reasonably achievable." In the case of liquid effluents, the total annual quantity of liquid effluent must be such that a calculated dose to an individual in an unrestricted area is less than 3 millirems to the whole body or 10 millirems to any organ.

13.9.1.21 Tritium Effluents and Releases (83)

The discussion of tritium evaporation during defueling is given in Section 6. Tritium in the form of water vapor would be released into the reactor building from the spent fuel pool and transfer canal. The evaporation rate would depend on the surface area of the pool and transfer canal (both of which are known) as well as on the humidity in the reactor building. The expected tritium concentration in the reactor building is not a health hazard to workers in the reactor building. Any release by purging to the atmosphere would be carefully monitored and kept within regulatory limits.

13.9.1.22 Data on Tritium Effluent Estimates (13)

The alternative of discharge to the river along with the radionuclide inventory of the processed water including tritum is presented in Section 7.2.

Dilution factors of 3400 are required for the radiological constraints and 3500 under the NPDES criteria. Since 34,000 gpm of dilution water from cooling tower blowdown is available and a total of 140,000 gpm of dilution capability is available from other equipment on site, discharge rates can be varied to satisfy Appendix I release limits at the discharge point, Primary Drinking Water Standards at a downstream uptake for drinking water and the NPDES limits. The dilution that satisfies the most limiting of these criteria will be used if this alternative is implemented.

13.9.1.23 Water Treatment at Intake (100)

The municipal water treatment processes at the point of potable water intake do not of themselves necessarily change the radionuclide concentrations in the river water. Should the alternative of controlled release of the processed accident water to the river be chosen, the control at the discharge is the most effective way to keep radionuclide concentrations at intakes below EPA Drinking Water Standards. This alternative along with others are fully described in Section 7.2. The staff concludes that processing alternatives exist such that if the processed water is permitted to be dispersed by controlled release into the river, the change in the radionuclide contents in potable water derived from downstream river water would be negligible.

13.9.1.24 Total Radionuclide and Concentration (100)

The PEIS presents concentrations of radioactivity measured in μCi/mL as well as the total radioactivity in curies. Both units are used, depending on which are most appropriate for the topic under discussion. For example, when discussing the efficiency of a cleanup alternative or environmental pathways to man, it may be more important to give the radioactivity concentrations. On the other hand, when discussing the radioactive waste liners and containers that have to be disposed of, the total radioactivity (Ci) may be a significant concern.

13.9.1.25 Cumulative Doses from Events at TMI-2 (100)

The impact of the accident can be found in numerous other reports assessing the accident, for example, NUREG-0558, "Population Dose and Health Impact of the Accident at the Three Mile Island Nuclear Station" published by the NRC in May, 1979. The radioactivity releases prior to the accident from TMI were well within the technical specification limits. When the licensee makes a cleanup proposal, the review of the proposal will include the potential cumulative health effects to the worker and the public which would result from that proposal. This PEIS has bounded these doses for the cleanup, which are summarized in Section 10.2 and 10.3.

13.9.1.26 Airborne Effluents (32, 55, 73, 83, 100, 116, 120)

The staff's estimate of the release of radioactive materials from cleanup activities to the environment considered several things, including: 1) the fraction of processed material that becomes airborne, and 2) the fraction of airborne material that passes through an air cleaning system without being collected. A brief discussion of the air cleaning system, including HEPA filters, is presented here to explain the staff's assumptions.

Tritium is an isotope of hydrogen that has chemical properties like hydrogen. Thus it combines with oxygen to form a molecule similar to that of H_2O, but that retains the radioactivity of the tritium. When the water is evaporated, all water molecules, including those with tritium, exist in a gaseous state. And, like other gases, they pass through a filter system. Other fission products (and actinides such as plutonium or uranium) exist in the process water as dissolved constituents. When the water is evaporated, the dissolved constituents stay behind in the remaining liquid. The evaporated water in the gaseous state contains none of the radioactive constituents and passes through a filter system unchanged. Evaporation is thus an effective method for separating bulk water from a solution containing dissolved radioactive constituents, but it is totally ineffective for separating tritium. This difference in behavior must be recognized in estimating radioactive releases.

However, evaporation, as it is conducted in plant operation, is not a perfect operation. Evaporation action creates turbulence at the surface which results in the formation of small droplets of water that are physically entrained in the rising vapor. The rising vapor contains only tritium, no dissolved radioactive materials; the entrained droplets contain both tritium and dissolved radioactive materials. In a well-designed evaporator, the fraction of the total vaporized material that exists in the form of entrained liquid droplets seldom exceeds 1×10^{-5}. This figure then forms the base for estimating how much dissolved radioactivity appears in the vapor stream. Following condensation, most of the entrained radioactivity appears in the condensate with a small quantity remaining with the uncondensed vapors. About 1×10^{-3} of the radioactivity in the condensate remains in the uncondensed vapors. Thus, less than 1×10^{-8} of the initial radioactive material remains in uncondensed vapor.

Other processing operations are generally less energetic and less turbulent than evaporation. In ion-exchange operations, process liquids are pumped in closed pipes through an ion exchange column. The fluids are relatively free of substantial turbulence. However, even in these operations, a fraction of the process liquids (less than for evaporation) may form small droplets that are physically entrained into surrounding airspaces, and are subsequently transported by the process off-gas system to the air-cleaning system. En route, the liquid droplets evaporate, leaving the dissolved radioactivity as a solid aerosol in the air stream. Entrainment for this kind of operation seldom exceeds 1×10^{-4} of the processed liquid and becomes airborne and enters the process off-gas streams. This value is conservative when compared with the information presented above and is compatible with data obtained from the processing of nuclear fuels. The latter consists of a series of many individual operations, some of which are far more complex than the few relatively simple operations envisioned for Three Mile Island.

The heart of the air cleaning system that pertains to the removal of particulate matter is the HEPA filter. In the PEIS, a penetration fraction of 1×10^{-3} was used for calculating effluents and releases thru the HEPA filters, a value that is conservative relative to achievable values.

The failure of a single HEPA filter was considered as a credible accident in the PEIS, that is why the staff requires two in series. The releases of a facility are constantly monitored with appropriate instrumentation. Failure of a HEPA filter would be evidenced by an increase in releases by the monitoring sytem and would alert operating personnel to take corrective action. The failure would have no effect on the functioning of the second filter in series with the first; thus containment integrity for the system would be maintained, but at a reduced level. The design of the monitoring sytem detectors is such that the failure of a single filter is detected within 15 minutes. No failures occurred during EPICOR-II processing of contaminated waters.

13.9.1.27 Potential for Airborne Releases from Fire (73)

The potential impact of fires considered to be even remotely credible has been addressed in appropriate sections of the PEIS. An accidental fire involving resins or spent fuel is not considered to be a credible event. The resins are contained in sealed steel liners, which are transferred to the onsite waste storage facility in a steel and lead cask. The storage facility consists of modules with thick concrete walls, and a 16-ton sealed concrete cover over each cell. The water is drained from the resins prior to storage, but they remain wet (like wet sand). In such a condition ignition and combustion is highly unlikely.

The nuclear fuel cannot sustain combustion, and the spent fuel is stored in water, with double containment used during transportation. Fire involving spent nuclear fuel is considered incredible under the conditions at TMI.

Combustible materials and fluids are controlled administratively to assure that nuclear fuel or resins will not be involved in a fire. In addition, fire control equipment is available on site.

13.9.2 Occupational Doses and Health Effects

13.9.2.1 Doses to Workers (4, 32, 78, 115, 116, 120)

Occupational doses for cleanup operations were estimated by the staff based on experience gained from work done in the past which include parameters such as:

- The manpower required to complete each task,
- The radiation field in which the work is likely to be done,
- The effectiveness of shielding and other radiations protection technique,
- Optimal use of personnel to minimize overall personnel exposure.

The estimates have been conservative, mainly due to conservative estimates of radiation field and manpower required to complete tasks. The estimating process has proven to be reasonably reliable for work done in the auxiliary and fuel handling building.

The auxiliary and final handling building exposure experience has been factored into the estimates for the reactor building.

The total radiation exposure through September 1, 1980, for the work effort in decontaminating the AFHB is given as 146 person-rem in Section 5.1.5.1. These values were obtained from analysis of personnel dosimetry records.

Current radiation protection standards for occupational exposure have been developed over the past 50 years. Most of the radiation standards were first proposed by the National Council on Radiation Protection and Measurements (NCRP). Other organizations which have undertaken evaluations of radiation exposure standards, particularly with respect to the cancer and genetic effects, include the National Academy of Sciences (NAS), the Medical Research Council (MRC) and the International Commission on Radiation Protection (ICRP). In January 1957, the NCRP recommended a maximum permissible dose of 5 rem per year for radiation workers. There is much information to show no effect on humans exposed to 1-15 rem per year. Therefore, it has been the judgement of the NCRP that the potential risk from a 5-rem per year occupational exposure is small. However, the NRC recognizes that the standards are based upon currently available information and should not be regarded as permanent. Accordingly, it has required that exposure to radiation be kept at the lowest practicable level when and wherever possible.

The licensee is required to ensure that the occupational doses that its radiation workers incur are as far below those specified in radiation protection regulations as reasonably achievable. Whole body exposures are monitored and recorded, and in situations where extremity (e.g., hand) doses can be higher than whole body doses, these extremity doses are also monitored and recorded. The licensee's radiation protection plan includes engineered safeguards and personnel access control to ensure protection from unnecessary and/or excessive inhalation of radionuclides. Respiratory protection is implemented whenever necessary. In this regard, air sampling is also performed to determine the level of any airborne contamination and document the adequacy of the engineered safeguards. In addition, provision is made for routine bioassay samples on an appropriate schedule for the radiation workers and prompt sampling and evaluation in the case of a suspected radionuclide intake. Blood cell counts are not generally taken because changes in cell counts occur only following relatively large over-exposures. The exposure rates associated with cleanup work at TMI-2 have not and would not be of this nature.

13.9.2.2 Radiation Levels for Workers After Shielding (50)

The estimated radiation levels are for worker exposure after precautions such as shielding and remote operations have been incorporated. However, these levels do not represent surface readings for filter cartridges or other components which may have localized areas of higher radiation.

13.9.2.3 Average Radiation Fields for Workers (50)

The 10 mR/hr dose rate is not an average used for all PEIS cleanup activities. It is, however, applied to all activities associated with reactor vessel head and internals removal and defueling. Based on conditions found in the containment building during the reactor building entries and decontamination achieved at other nuclear facilities, when the reactor building sump and basement have been drained and limited cleanup completed, average exposure levels of 10 mR/hr are believed to be achievable. Average radiation fields are difficult to estimate; however, there is a general tendency to over-estimate.

13.9.2.4 Radiation Protection Plan (115)

A radiation protection plan to ensure radiation exposure to cleanup workers are below regulatory limits and to keep those exposures as low as reasonably achievable (ALARA) has been submitted by the licensee to the NRC staff for review and approval.

13.9.2.5 Radiation Dose Estimates to Workers for Cleanup Alternatives (108)

Potential occupational radiation exposures to cleanup workers are evaluated for every cleanup alternative discussed in the PEIS. The potential health effects of worker exposures are summarized in Section 10 of the statement.

13.9.2.6 Airborne Radioactivity Levels During Water Processing (55)

Water processing would take place inside enclosed systems. Although the airborne effluents prior to treatment by HEPA filters may exceed MPC levels, these effluents will not be exposed to building air and should not cause the airborne radioactivity levels outside of the system to exceed MPC levels.

13.9.2.7 Worker Exposure Estimates for Reactor Coolant Sampling (32)

Worker exposure is estimated to be 20-30 mrem per sample while taking reactor coolant samples. It is recognized that during the accident, taking coolant samples resulted in some overexposures. However, the radioactivity levels of reactor coolant are now orders of magnitude lower than those during the accident because of substantial decays of shorter half-life radioisotopes. Other factors also contributed to the high exposures when the samples were taken during the accident, e.g., urgency of the situation, high area radiation levels, lack of radiation shielding and proper radiation protection procedures. These conditions which existed during the emergency sampling have been corrected and would not be expected to recur during the cleanup.

13.9.2.8 Occupational Dose Records (69, 114, 115)

Questions were raised regarding records of occupational doses. The NRC requires all its licensees to maintain complete radiation exposure records of its employees. Whenever an employee joins a licensee, the licensee is required to obtain a record of the new employee's exposure history. Such a record is usually obtained by requesting exposure records from the employee's former work places. At TMI, the cumulative occupational dose information is entered into the Radiation Work Permits of each cleanup worker for entry to radiation areas. The licensee has a computer program to record the cumulative occupational doses. This information is updated by the TLD dosimeter readings of each cleanup worker and is maintained for both regular and transient workers on a permanent basis. Nonoccupational doses, e.g., medical exposures, are not recorded.

13.9.2.9 Cancer Rates and Exposures for Workers (125)

The staff uses well documented and accepted risk estimators in evaluating health effects. The staff estimates of the health effects occurring to the occupational work force, (the largest environmental impact identified) would indicate a very small increase in the cancer rate that would normally be experienced due to causes other than occupational radiation exposure.

13.9.2.10 Doses to Workers and Effect on Gene Pool (67)

Radiation protection standards for radiation workers are higher than for the general public. However, radiation workers are a small fraction of the population, so their contribution to the total genetic pool is proportionately small. Radiation standards are designed to limit any increase in overall mutation rate in the total genetic pool to very small fractions of the normally occurring rate. Since radiation workers make up only a small fraction of the general population, radiation doses for them can be somewhat higher than for the general population without causing a large increase in overall mutation rates. It should be noted that many industrial processes other than work involving radiation also expose workers to mutagens.

13.9.2.11 Health Effects from Doses to Workers (34, 50)

In section 10.2.2, the estimate of 131 fatal cancers per one million person-rem is for people in the worker age group. The cleanup workers involved constitute the assessed population. The 135 fatal cancers per one million person-rem is based on a lifetime in the general population.

13.9.2.12 Occupational Doses Due to Tritium (79, 85)

Processed water containing tritium has been used in decontamination at TMI-2 to reduce the volume of tritiated water that must be stored for eventual disposal. There is an occupational exposure associated with the use of the tritiated water.

Prior to the use of processed (tritiated) water in the water jet for the auxiliary and fuel handling building decontamination, tests were conducted by the licensee to determine the tritium level in the atmosphere. These tests were done with little ventilation and indicated that the tritium level in the atmosphere during the use of the water jet was approximately 10% of the allowable airborne concentration (10 CFR 20). The water jet tests are expected to be bounding on airborne tritium concentrations; other activities, such as wet mopping or hose washing, are expected to have lower airborne concentrations. During future activities, monitoring of airborne tritium will be done as appropriate.

When compared to the estimates of the occupational dose due to the radiation fields produced by the contamination on the surfaces, the dose from tritium would be a small increment in the estimate of the occupational dose incurred during the decontamination efforts.

13.9.2.13 Radiation Exposure from Waste Storage Facility (100)

The exposure rate indicated in the PEIS for the waste storage facility is less then 0.5 mR/hr at the facility boundary, not the site boundary. The site boundary and personnel working areas are well removed from the waste storage facility boundary. The radiation levels at the site boundary will be maintained within regulatory limits.

13.9.2.14 Effect of Schedule on Occupational Exposures (115)

An extension of schedule in itself does not necessarily imply that the level of effort has increased, or that personnel doses or total dose will increase. An extension in schedule may have many causes and those that have occurred at TMI have generally been because of the need for licensing approvals and extra time spent to assure that the best approaches are used. Problems have been encountered which have caused delays, but these have not generally increased total or individual worker exposures. In fact, even though manpower estimates have increased, experience at TMI has shown that exposure values have generally compared well with predictions.

13.9.3 Offsite Doses & Health Effects

13.9.3.1 Offsite Dose Criteria (20, 85, 52, 80, 64)

10 CFR Part 50, Appendix I requires that reactors be designed so that the annual dose commitment to the maximum exposed individual offsite does not exceed 5 mrem total body or 15 mrem skin dose for gaseous releases. Dose predictions made prior to the Kr-85 purge indicated that the Appendix I design objectives would not be exceeded during the purge. Monitoring during the purge process confirmed that the dose predictions were conservative. 10 CFR Part 50, Appendix I also requires that reactors be designed so that the annual dose commitment to the maximum exposed individual offsite not exceed 3 mrem for liquid pathways. Dose estimates made in this PEIS indicate that over the course of the entire decontamination process, including the discharge of processed water to the river, if proposed and authorized, feasible alternatives exist such that the Appendix I dose commitment through liquid pathways will not be exceeded. Criteria regarding offsite doses recommended for the cleanup operation are described in Section 1.6. These criteria will assure that the releases are made in such a manner to conform to the Appendix I design objectives.

13.9.3.2 Offsite Dose to be Kept as Low as Reasonably Achievable (ALARA) (52, 109)

The doses that will occur as a result of the cleanup operation are summarized in Table 10.3-1. The numeric values in this table are believed to be conservative and represent upper bound values. The basis for this conservation is described in Appendix W of the PEIS. As described in the PEIS, the basis used in approving programs will depend partly on the success of earlier programs in meeting the ALARA principle (10 CFR 20). Further, the licensee will be required to periodically report to NRC estimates of offsite doses resulting from the decontamination operations. If a program is resulting in offsite doses significantly larger than originally estimated, the NRC can require modification as appropriate.

13.9.3.3 Dose Estimates for Future Cleanup Alternatives are Projections (20)

The dose estimates in the PEIS for future cleanup alternatives are projections because the releases have not yet been made. Actual measurements will be made when releases occur. A summary of the projected estimates and the associated potential risk probabilities can be found in Section 10 of the statement.

13.9.3.4 Comparison of Cleanup Impacts with Other Sources (116)

Section 10.3 provides additional information on the comparison of risks of the cleanup with other types of human activity.

13.9.3.5 Comparison of Cleanup Doses to Background Radiation (99, 100)

The potential offsite radiation doses resulting from cleanup activities and the naturally occurring background radiation were compared to illustrate the relative significance of radiation doses resulting from the cleanup. It was suggested that the comparison should be made because of the different exposure pathways and types of radiation involved. Potential health effects from whole body dose or dose to certain organs are independent of the pathway. The dose conversion factors used in the assessment were based on the different radioisotopes and types of radiation involved. The result was an equivalent effective dose value.

13.9.3.6 Dose to Maximumly Exposed Individual Considered (60)

The dose estimates discussed in the PEIS represent the potential dose to the maximumly exposed individual, as well as the average dose to the population from the cleanup alternatives. A discussion of the models and the methods used to calculate these values are provided in Appendix W to the PEIS.

13.9.3.7 Offsite Locations and Radiation Doses (67)

The offsite dose calculation of a few millirem as a result of all cleanup activities is estimated for the individual residing at the most critical location. It has been suggested that the farther away from the TMI site, the better off the individual may be. This, however, needs to be qualified, because it would depend on where the individual goes. Natural background radiation levels in the U. S. vary considerably (from 70-310 mrem/yr), and in many locations in the U. S., the background radiation would be higher by more than the few millirems which would result from the cleanup plus the natural background radiation (116 mrem/yr) at the critical location near TMI.

13.9.3.8 Offsite Dose Fluctuation Ranges (32)

The actual offsite monthly dose rate could be different from the values calculated because the calculations are based on annual average conditions. The differences would not be expected to be greater than a factor of "four" which is acceptable for these low doses.

The background radiation levels reported in the PEIS do not include man-made radiation exposures, such as radiation exposures due to medical diagnosis. In addition, it has been found that the radioactivity releases due to nuclear power generation are not significant enough to impact on the average range of background radiation levels reported.

13.9.3.9 Offsite Contamination from Cleanup (37)

The NRC appreciates the concerns of people who live in communities around TMI, and agrees that the benefits of nuclear power would be questionable if nuclear waste did result in widespread contamination of farm lands or cities. However, water, air, and vegetation are being monitored and there is no evidence of contamination which would cause radiation doses which are more than a very small fraction of the dose from natural background radiation (due to natural radioactivity in soil, water, air, the human body and from cosmic radiation). Also, instrumentation is available with which radioactivity can be traced and measured, so contamination in the environment would be detected and mitigative actions taken long before it could become a serious problem.

13.9.3.10 Drinking Water Standards and Immersion Dose (67, 72, 79)

In addition to the reference to Drinking Water Standard, the cleanup operation also has to meet the numerical design objectives in Appendix I to 10 CFR Part 50, which state that the dose to an individual offsite should not exceed 3 mrem total body through all liquid pathways. This would include the dose from bathing and other use of the Susquehanna River for recreational purposes. In the PEIS, the dose estimates consider all the major pathways.

The assessment in the PEIS is based on calculations of radioactivity concentrations in the water and the potential radiation dose to the maximumly exposed offsite individual. The conclusions in the PEIS are based on these dose pathway calculations, and not on the EPA Drinking Water Standard, although meeting the standard would be a condition to be considered for processed water release.

13.9.3.11 Calculation of Dose from Drinking Water (50, 52, 55, 85)

Doses are calculated for the nearest downstream drinking water intake at PP&L's Brunner Island Station approximately five miles downstream. An average river flow of 12,600 cfs is used for dilution. Calculation models and parameters used in estimating these doses are discussed in Appendix W. The assumption that the effluents will be well mixed at the drinking water intake is considered to be realistic.

13.9.3.12 Dose Rate at Site Boundary (52)

Proposed criteria to adopt Appendix I would require offsite doses due to airborne releases to be less than 5 mrem/yr whole body dose and less than 15 mrem/yr skin dose. Doses due to liquid pathways would have to be below 3 mrem total body dose. The direct radiation level at the fence surrounding the interim storage and staging facility will be less than 0.5 mrem/h. The direct radiation level at the site boundary will be substantially smaller because of decrease due to distance.

13.9.3.13 Tritium Dose through Liquid Pathways (66)

Tritium has its primary importance in the drinking water pathway as opposed to the fish consumption pathway because the fish only equilibriate with the tritium concentrations in the water and do not concentrate this radionuclide. The concentrations of tritium in the fish would be the same as in the drinking water.

13.9.3.14 Errors in Drinking Water Calculations in Draft PEIS (76)

The staff has completely revised the calcullations on drinking water doses, incorporating new information made available by the licensee and other sources. These calculations are indicated in Appendix W and the results are indicated in Section 7.2.

13.9.3.15 Environmental Impact During Normal Power Operation (46)

The staff concludes in this PEIS that the cleanup at TMI Unit 2 could be accomplished with offsite radiation doses to the public no greater than those assessed for the normal operation of TMI-2 prior to the accident. The environmental impacts associated with the normal operation of the TMI Unit 2 are described in "Final Supplement to the Final Environmental Statement Related to the Operation of Three Mile Island Nuclear Station, Unit 2," (NRC report NUREG-0112, December 1976).

13.9.3.16 Tritium Release Due to Reactor Pressure Vessel Head Removal (76)

The offsite dose from tritium released from the reactor vessel during head removal would be extremely small, much smaller than that now occuring whenever the reactor building is vented, because the sump water constitutes a much larger source of tritium. Special precautions against tritium release during head removal would not be warranted.

13.9.3.17 Offsite Dose from TMI-2 Accident Taken Into Consideration (70, 93, 109, 114, 67, 128, 20, 85, 84)

The dose and potential health impacts that occurred as a result of the accident and the methodology behind these dose calculations are contained in NUREG-0558, "Population Dose and Health Impact of the Accident at the Three Mile Island Nuclear Station." The offsite doses from the cleanup operation will be negligible (few mrem) in comparison to that from the accident (less than 100 mrem).

13.9.3.18 Basis for Dose Estimates to Base on Major Isotopes (4, 73, 84)

Appendix J presents the basis for considering only a few of the radionuclides present (is H-3, Cs-137, Cs-134, Sr-90, and Sr-89) in the source spectrum for offsite dose calculations. By use of the existing concentration in the facility and MPC values of 10 CFR Part 20 (which incorporate usage factors and ingestion dose conversion factors for water, and inhalation rates and inhalation dose conversion factors for air) the relative significance from the standpoint of dose of each nuclide in the entire spectrum is estimated.

13.9.3.19 Justification of Dispersion Factors in Dose Calculations (100)

For the application to operate the Unit-2 facility, the licensee was required to establish both long-term and short-term meteorological dispersion data. These data were reviewed by the NRC staff and described in the Final Environmental Report and Safety Evaluation Reports related to the operation of the TMI-2 facility. The NRC staff has also determined that meteorological conditions presently existing in the vicinity of TMI-2 do not differ significantly from those described in those reports.

13.9.3.20 Conservatism in Dose Rate Estimates (52, 55, 84, 107)

The decontamination program is scheduled over a period of several years, hence the approach used in the Draft PEIS of evaluating the total release as if it occurred over a period of one year actually overestimates release rates and is overly conservative. Meteorological parameters specific to accidents are now used for these calculations in the PEIS (see Appendix W).

13.9.3.21 Radionuclide Pathways to Humans (72, 68)

The nuclides which could be released would accumulate in the body through inhalation of the plume, and by ingestion (drinking water, eating fish, meat and farm produce, and drinking milk, etc.) These pathways are considered in the dose calculations.

13.9.3.22 Dose Commitment to Humans Through Aquatic Food Chain (67, 13, 55)

All the doses that were calculated in the PEIS include accumulation in the food chain in the final receptor next to man, e.g., fish and shellfish. Also, all of the internal doses that were calculated for human beings includes the "50 year dose commitment effect." This means that the one-time intake includes the dose that is received from the one-time intake out to a period of 50 years.

13.9.3.23 Meteorological Dispersion Factor (55)

The value of meteorological dispersion factor (X/Q) which was used throughout the text for estimating maximum individual doses due to gaseous releases during normal cleanup activities was 6.7×10^{-6}. This X/Q value represents the highest annual average X/Q value for all the 16 sectors at the site boundary.

13.9.3.24 Estimates Made for Organ Dose (32, 55, 121)

The PEIS contains estimates of critical organ doses and their associated health effects.

13.9.3.25 Offsite Radiation Doses by Inhalation Considered (120)

People will absorb radioactive materials such as Sr-90 and Cs-137 by breathing if they are downwind of the plant and these materials are being released to the atmosphere. The dose calculations of the PEIS include this pathway.

13.9.3.26 Dose Factor per Unit Activity Released (73)

The population dose estimates indicate that for every curie of Sr-90 released uniformly over one year to the atmosphere, the resulting 50-mile population dose is about 53 person-rem. For each curie of Cs-137 and Cs-134, the 50-mile population dose is 33 person-rem and 40-person rem, respectively.

13.9.3.27 Tritium Dose Considerations (66, 114, 115, 117, 130)

Tritium behaves differently from other isotopes in that it becomes distributed in the body in a manner that is proportional to the water content for each organ. As a result, except for bone, the dose that each type of organ tissue receives is about the same for a given amount of activity taken in by the body. Other isotopes generally have varying affinities for the different body organs. Regarding its effect on tissues, tritium is a weak (18 Kev) beta emmitter, therefore, its effect will be fairly local and inter-organ effects are small to negligible.

Tritium can often be a problem from the engineering standpoint because it is chemically indistinguishable from the hydrogen in water. Therefore, it is not removed in filtering or ion-exchange systems. For this reason it is usually given special consideration, rather than because of any special behavior within the body. Except for its uniform distribution within the body, it is not expected to behave any differently from the standpoint of dose effects than any other beta emitter.

13.9.3.28 Exposure to the Public from Waste Disposal Operations (107)

The dose to people living near the disposal sites has not been calculated for specific TMI waste shipments since the TMI waste does not add to the waste disposed of at the facility; rather it replaces other waste shipments. Thus, the TMI waste does not add to the exposure received by the population in the vicinity of those facilities.

The performance objectives for a waste disposal facility will be (1) doses to inadvertent intruders will not exceed 500 mrem/yr and (2) doses from groundwater pathways will not exceed 25 mrem/yr at the disposal site boundary.

13.9.3.29 Global Dose Commitments (120)

In the PEIS the 50-year dose commitments were estimated. This is generally conservative as the average age of the human population is over 20 years and the life span of a person is 70 years. Estimating dose commitments beyond this does not realistically reflect the age structure of the population. Extrapolating population doses beyond 50 miles is meaningless because they are so small that not only are they greatly exceeded by natural background doses, but they are exceeded by fluctuations in natural background doses.

Calculations of global doses due to releases at TMI serve no useful purpose as the all-time background dose to the global population is infinite. If it is assumed that the 13 person-rem dose is received by the 4×10^9 people who are now alive, each person would receive an average dose of 3.2×10^{-6} mrem. A dose of this value has no meaning.

13.9.3.30 Radionuclide Concentrations and Population Doses (73)

Population doses are directly related to activity that is released. In fact, individual doses are also directly related to activity. Nevertheless, both doses are inversely related to factors related to dilution; for example, the river flow. The calculational models used by NRC work exactly in this manner. The assumptions described in the PEIS indicate that both activity and river flow are used in the calculations. There are at least three reasons for discussing dilution and concentrations: First, there are federal regulations requiring that certain concentrations not be exceeded. Second, the greater the dilution, the lower the dose rate. Low dose rates are desirable because basic health physics principles suggest that, for the same cumulative dose, biological tissue is less affected by lower dose rates than higher ones. Third, the closer that concentrations and dose rates are to background concentrations and dose rates, the less chance the release can perturb anything in the ecosystem. The ideal situation is to keep release levels within the natural fluctuations of the ecosystem. The approach that is used in discussing these matters in the PEIS is not intended to create false impressions, rather, it is to present the releases from a perspective that the majority of the public will understand. Comparing doses to background, and concentrations to federal standards, seems a reasonable approach.

13.9.3.31 No Basis to Declare Region Around Unsafe (73)

The potential environmental impacts of cleanup alternatives were evaluated in the PEIS. Based on the potential environmental impacts discussed in the PEIS, there is no basis for declaring the region around TMI unsafe for human habitation nor that agricultural products from the area be declared unfit for human or livestock consumption.

13.9.3.32 Too Many Units Used for Radioactivity (121)

It is necessary to use several different units in the PEIS when referring to radiation since different aspects of radiation are being treated. The units which describe these aspects have been adopted by the International Commission on Radiation Units and Measurements (ICRU) Report 33 - Radiation Quantities and Units.

13.9.4 Health Effects from Radiation Exposure

13.9.4.1 Health Risks Associated with Radiation Doses During Cleanup (2, 20, 59, 60, 67, 69, 82, 91)

Several studies analyzed the data relating to the influence of the magnitude and temporal distribution of dose on the biological effectiveness of low linear energy transfer (LET) radiation per unit absorbed dose. All of the isotopes that could be released as a result of cleanup activities are of this category. The National Council on Radiation Protection and Measurements (NCRP) concluded in Report No. 52,[1] "that there has been no direct demonstration of deleterious effects from worldwide fallout and therefore from environmental Cs-137. While this lack of evidence does not prove a total absence of damage associated with internal exposure to Cs-137, it does indicate that risks due to present and past levels of environmental contamination of Cs-137 have been at levels so low that harmful effects, if they exist, are not readily apparent." Also, in the summary and conclusion of BEIR-III Report[2] the committee concluded that: "It is unlikely that carcinogenic and teratogenic effects of doses of low LET radiation administered at this dose rate (100 mrad/yr) will be demonstrable in the foreseeable future." Furthermore, it is likely that the risk models from low-level radiation are conservative, in fact, certain studies with animals suggest the existence of a threshold below which there are no adverse health risks. Hence, it may be that there are no health effects, detrimental or otherwise associated with these releases.

The PEIS describes health risks to the plant workers and to offsite individuals and population. The regulations, 10 CFR Part 20, require that an individual plant worker receive no more than an average of 5000 mrem per year occupational exposure. This is equivalent to a lifetime chance of 7×10^{-4} for the worker to die from cancer as a result of the radiation exposure based on calculational methods described in the PEIS. The risk of mortality from lung cancer due to smoking one pack of cigarettes per day is 5×10^{-3} per year and the risk of mortality from driving an automobile 10,000 miles per year is 7×10^{-5} per year. There are other examples that could be used but these are sufficient to illustrate that the greatest threat to plant workers is not from radiation exposure.

13.9.4.2 Up-to-Date Information on Health Effects (99)

At the time the draft PEIS was written, the Committee on Biological Effects of Radiation had not released its 1980 draft report (BEIR-III)[2]. However, in the Summary and Conclusions of BEIR-III, the committee concluded that low-level radiation derived from the earlier BEIR reports are probably conservative.

13.9.4.3 Epidemiological Study on Low-level Radiation (10, 60, 64, 72, 107)

In a study funded by the NRC and reported in NUREG/CR-1728, the feasibility of epidemiologic investigations of the health effects of low-level ionizing radiation was investigated. In the summary to NUREG/CR-1728, it was concluded that occupational groups are the most practical to study and that it is not feasible to study the dose-response effect for cumulative doses that are less than 10,000 mrems. The maximum offsite dose resulting from the accident was about 100 mrem. The annual maximum dose from TMI-1, should it be restarted, will be about 10 to 20 mrem, and the maximum dose from the decontamination program will be a few millirems. If TMI-1 operates for

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[1] National Council on Radiation Protection and Measurements, "Cesium-137 from the Environment to Man: Metabolism and Dose," Report No. 52 (1977).

[2] Committee on the Biological Effects of Ionizing Radiation. "The Effect on Populations of Exposure to Low Levels of Ionizing Radiation," National Academy of Sciences, 1980.

30 years, the cumulative offsite maximum dose will be about 300 to 600 mrem. The cumulative 30-year dose from the accident, operation of Unit 1, and the decontamination program is 400 to 700 mrem, a number much smaller than the 10,000 mrem lower limit study feasibility number. Hence, a detailed epidemiological study around TMI would not likely detect any health effects that could be related to doses from TMI. NRC is not requiring the licensee to do these types of studies.

13.9.4.4 Detection of Radiation Health Effects (67, 72, 128)

The amount of radioactivity that will be released will be so low that no effects will be observable. The radioactivity released will result in doses of a few millirems to ten millirems over the entire decontamination period (several years in length). The average individual doses for the 50 mile human population will be much smaller than these maximum individual calculations. Natural background radiation is about 116 mrem per year in the Harrisburg area. One study suggests that doses of 10,000 mrem would be required before epidemiological effects could be detected in humans.[1]

13.9.4.5 Approach to Estimate Health Effects (67, 121)

To estimate the health effect due to cleanup alternatives the PEIS adopts the "linear no-threshold" approach. Thus the health effect is estimated to be linearly proportional to the dose received. That is, doubling the dose would double the probability of potentially harmful health effects. Also, there is no threshold dose below which there is zero probability of potential health effect. Therefore, although the radiation dose to the public from cleanup alternatives is very small, in the order of a few millirems, a correspondingly small probability of health effect still exists. This is in the order of less than one in one million.

13.9.4.6 Risk Estimators for Radiation Doses (73, 98, 114, 125)

Just as there are studies suggesting that risk estimators should be higher than the ones that were used in the PEIS calculations, there are also studies suggesting that they should be lower. For example, people who live in Colorado, Wyoming, and New Mexico receive considerably more background radiation than the average U. S. citizen. Nevertheless, their leukemia rates are lower than that of the average U. S. citizen.[2] The risk estimators that were used in the PEIS were based on the BEIR-I report.[3] The latest BEIR report (BEIR-III) suggests that the risk estimators should be even smaller than those of BEIR-I.

13.9.4.7 Accuracy of Dose-Risk Estimates (50)

Cancer risk estimates are not accurate to 20%, therefore rounding off is justified. The meaning of cancer risk estimates was described in BEIR-II in the following manner: "As suggested by the ICRP, the expression of risk estimates in absolute terms--for example, 2 cases per one million exposed people per year per rem--might be misinterpreted as implying considerably greater accuracy than the facts justify. For this reason, estimates are sometimes expressed in terms of 'orders of risk,' e.g., 1 to 10 cases/10^6/year/rad is a 6th order risk."

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[1] The Feasibility of Epidemiological Investigations of Health Effects of Low Level Ionizing Radiation," NRC Report NUREG/CR 1728, Nov. 1980.

[2] Cohen, Bernard L., "The Cancer Risk from Low-Level Radiation," J. Health Physics, Vol. 39, pp. 659-678, 1980.

[3] Report of the Advisory Committee on the Biological Effects of Ionizing Radiations, "The Effects on Population of Exposure to Low Levels of Ionizing Radiation," National Academy of Sciences, p. 99, November 1972.

13.9.4.8 Lifetime Health Effect Estimated (67)

The dose estimates given in the PEIS are the total doses that the maximum individual could receive during the entire decontamination process. This means that if the decontamination lasted longer than one year, the total dose received would be the same as if the decontamination lasted only a year. The risk estimates stated in the PEIS are the lifetime risks based on a one-year exposure dose.

With regard to the data base used to determine risk estimates in the PEIS, most of the data came from human beings. These estimates are based on studies of the survivors of the Hiroshima and Nagasaki bombings, and also from other individuals exposed occupationally, e.g., in uranium mining or as a result of medical diagnosis or therapy. With the Hiroshima and Nagasaki populations we are able to observe a wide spectrum of ages, from exposed fetuses to older people. In this manner, the dose effect on all ages are taken into consideration.

13.9.4.9 Offsite Health Effects To Be Essentially Non-existent (120)

Risk comparisons in the PEIS are made in an attempt to quantify the impact. What the PEIS suggests is that the risks associated with this decontamination program are within levels considered normal. The statement that health effects are essentially non-existent is based on a comparison of the calculated doses to background. This is not inconsistent with the quantitative estimates made farther on, because the quantitative estimates result in negligible risk.

13.9.4.10 Controversy of Health Effects at Low Radiation Levels (64)

The low levels of radiation which may be received by individuals living close to the plant were estimated to be from one to a few millirem. There are no risk models suggesting that this level of radiation is harmful. The controversy exists at much higher levels of radiation and in calculating population risks from very low radiation levels. Nevertheless, the individual risk is insignificant.

The following table, which lists the sources of radiation and associated doses for a typical individual living on the east coast, illustrates the insignificance of a dose of a few millirem.

Sources of the Average East Coast American's
Annual Intake of Radiation

Source	Dose (mrem)
Natural	
Sky	35
Housing	34
Food	25
Ground	11
Air	5
Total Natural Sources	110
Man-made	
Medical and Dental	41
Weapons Fallout	4
Total Man-made Sources	45
Total All Sources	155

"Nuclear Radiation and Health" by Roger E. Linnemann, M.D.;
The Environmental Impact of Electrical Power Generation:
Nuclear and Fossil" by Pennsylvania Department of Education.

The quantities in the table can vary significantly from one individual to the next. For example, the sky dose depends on the amount of time spent outdoors, the housing dose depends on the type of construction materials used for the house, the food dose depends upon the quantity of food ingested, and the medical dose depends on the amount and type of treatments an individual receives. The variations are far above the doses that could occur as a result of the cleanup operation.

13.9.4.11 Synergistic Effects and Health Effect Estimates (121)

The potential increases in fatal cancer assessed in the PEIS does take into account synergistic effects in the same way as the normal cancer death rate of 1 in 5. We cannot state exactly what the synergistic effects are in either case, but since the risk estimators used are based on real population groups, the synergistic effects, if any, are correctly factored in.

13.9.4.12 Potential Health Risks Estimated (111)

The PEIS includes estimates of potential doses and associated health effects and environmental impacts. From these analyses, the staff concludes that the cleanup can be accomplished by using alternatives having such radioactivity release levels that potential effects on the health of the public and the safety of the environment would be negligibly small, if not zero.

13.9.4.13 Risk Factors for Non-fatal Cancer (73, 121)

The risk estimators used in PEIS represents rate of lifetime cancer mortality per unit radiation dose. These risk estimates were developed in WASH 1400, Appendix VI. Appendix III of WASH-1400 provides an estimate of non-fatal cancer risks. The risk for non-fatal cancers is about 1.5 times that of fatal cancers. The PEIS concludes that no significant health effects and risks were expected as a result of decontamination of TMI-2. This conclusion would have been the same if non-fatal cancer risks were included.

13.9.4.14 Hospital in Vicinity to Treat Radiation Patients (99)

It is current NRC policy that there be at least one hospital in the vicinity of a nuclear power plant that is able to treat patients either contaminated with radiation or suffering from overexposure to radiation. In the case of TMI it is the Hershey Medical Center in Hershey, Pa.

13.9.4.15 Natural Background Radiation (22, 52)

The natural background radiation includes sources from cosmic and terrestrial radiations. It does not include man-made sources such as radiation from fallout, medical and dental X-rays. Nuclear power operation radioactivity releases are not significant enough to impact on the natural background radiation used in the PEIS.

13.9.5 Radioecological Effects

13.9.5.1 Radiation Doses and Effects on Ecological Community (46, 59, 68, 75, 117, 130)

Doses to animals in the vicinity of Three Mile Island are not expected to be significantly greater than that received by humans. Although guidelines have not been established for acceptable limits for radiation exposure to species other than man, it is generally agreed that the limits established for humans are also conservative for other species. Experience has shown that the maintenance of population stability is crucial to the survival of a species; species in most ecosystems suffer rather high mortality rates from natural causes. While the existence of extremely radiosensitive biota is possible, and while increased radiosensitivity in organisms may result from environmental interactions with other stresses (e.g., heat, biocides, etc.), no biota have yet been discovered that show a sensitivity (in terms of increased morbidity or mortality) to radiation exposures as low as those expected in the area surrounding the Three Mile Island nuclear power plant. Furthermore, in all the plants for which an analysis of radiation exposure to biota other than man has been made, there have been no cases of exposures that can be considered significant in terms of harm to the species, or that approach the exposure limits to members of the public permitted by 10 CFR 20.[1] The BEIR Report[2] (which is still being reviewed by the NRC staff) concluded that the evidence to date indicates that no other living organisms are very much more radiosensitive than man. Therefore, no measurable radiological impact on populations of biota is expected as a result of the decontamination operation of this plant.

Long-term bioaccumulation is generally not considered as important in animals as in humans because humans generally have the greater longevity. Thus long-term problems are most significant in humans. The dose calculations for humans represent the 50-year cumulative dose based on a one-year intake of radioactivity.

13.9.5.2 Studies of Plants and Animals (64, 88, 123)

In Section 3.1.5 in the PEIS is a summary description of the ecology of the Three Mile Island vicinity and other areas potentially affected during cleanup operations. In-depth specific accounts of the aquatic ecology and fisheries of the Susquehanna River and Chesapeake Bay are included in Appendix E. Appendix E was prepared from an extensive review of the literature, with the most pertinent sources cited and included in the references.

No experimental research per se on the ecosystems of the TMI vicinity has been done specifically for the PEIS. However, studies of the aquatic ecology and fisheries of the Susquehanna River near TMI have been ongoing continuously since 1974. These studies give us the description before the accident. These are summarized in PEIS Section 11.10.

The NRC staff has carried out post-accident examinations of plant and animal health effects in the TMI area. The results have been reported in two NRC technical reports. NUREG-0596 examined disease, parasites, abnormalities, and mortality of fish in the Susquehanna River following the accident and compared the figures with data for corresponding time periods during the pre-accident years of 1975-1978, and with data from other upstream and downstream areas of the river. NUREG-0738 investigated many reported health problems with plants, animals, and livestock following the accident. That report summarizes the investigations of terrestrial biologists, veterinarians, and a radiobiologist.

Studies of aquatic and terrestrial populations near TMI are continuing under the NRC Environmental Technical Specifications. The aquatic biological and fisheries studies are summarized in Section 11.10 and in Appendix E. The detailed studies of the river biota occur in the York Haven Pond near TMI. Sampling points in the pond are upstream and downstream of TMI and in all three river channels.

The studies have documented no ecological or animal health impacts from the accident. The studies do not include areas "indisputably" beyond the range of accident effects (had there been any). To include such locations could mean sampling in areas far removed (i.e., several miles upstream) from TMI where the biotic community and its habitat are different from those control stations. However, data from the post-accident period and from the ongoing cleanup period have been and will continue to be compared with comparable data collected during the 5 year pre-accident period of 1974-1978. Those pre-accident ecological and animal health data serve as baseline "control" information against which to compare similar data during cleanup.

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[1] Blaylock, B. G. and Witherspoon, J. P., "Radiation Doses and Effects Estimated for Aquatic Biota Exposed to Radiative Releases from LWR Fuel-Cycle Facilities, Nuclear Safety, Vol. 17, No. 3, May-June 1976.

[2] Committee on the Biological Effects of Ionizing Radiation. "The Effect on Populations of Exposure to Low Levels of Ionizing Radiation," National Academy of Sciences, 1980.

13.9.5.3 Effects on Ecology of River and Bay (11, 20, 46, 79, 91, 99, 103, 121, 123)

The potential impacts to the ecology of the Susquehanna River and Chesapeake Bay are discussed in detail in Section 7.2.5.4 of the PEIS. The staff carefully considered potential biological concentration factors as well as dilution and dispersal of the radionuclides, bioaccumulation factors, and the radionuclides in sediments when evaluating the potential impacts to the river of the radionuclides released in the processed water. The staff's original conclusion that the release of processed water is not expected to have adverse environmental impacts is still valid.

It is not true that any input to an ecosystem must have an irreversible effect. Indeed, it is one of the basic tenets of ecology that ecosystems exhibit homeostasis and are capable of self-maintenance and self-regulation if perturbations (such as input of pollutants) are not too severe (E. P. Odum, 1971, Fundamentals of Ecology, 3rd ed., W. B. Saunders Co., Philadelphia).

The discharge of processed water to the Susquehanna River is not "assumed" but is discussed as one alternative means of disposing of the water containing tritium, which is the only radionuclide of consequence. There are discussions of accidents which, if they should occur, would result in processed or unprocessed water being released to the river. Concentrations of radionuclides in fish and shellfish as a result of both controlled releases of processed water are discussed in Section 7.2.5.4 of the PEIS. The biota of the Susquehanna River and the Chesapeake Bay would receive a small fraction of the radiation dose received from natural background, and the incremental effect would be unmeasurable.

Marketability of seafood products from the Chesapeake Bay is also discussed in Section 7.7.5.5.

13.9.5.4 Bioaccumulation Factors in Fish Considered (130)

The bioaccumulation factors are factored into the calculations of radioactivity in fish flesh and the doses to fish. The approach that is used in calculations is one of conservative averaging. For example, the bioaccumulation factors selected are the ones believed to be conservative. In addition, conservative flows were used and no credit is taken for sedimentation.

The overall modeling approach has been used at many reactors. Follow-up field studies confirm that the approach is conservative as a whole. This is not to say that a given individual fish may not have a greater nuclide burden than the calculated value, but only that it is very unlikely that the average value exceeds the calculated value. Estimates of bioaccumulation of strontium and cesium in fishes of the Susquehanna River based on actual field data are given in Section 7.2.5.4.

13.9.5.5 Bases for Bioaccumulation Factors (13, 72, 91, 100)

The rationale for bioaccumulation factors used by NRC is given in a report by Stanley Thompson et al., "Concentration Factors of Chemical Elements in Edible Aquatic Organisms," Lawrence Livermore Laboratory, October 10, 1972. For strontium uptake in freshwater fishes two references are cited. One reference indicated an average concentration factor of 3.0 and another indicated one of 16. For cesium uptake in freshwater fishes six references are cited with values ranging from a low of 390 to a high of 4700 with a weighted average slightly greater than 2000.

The bioaccumulation factors for Sr and Cs have been observed to be greater for freshwater fishes than for saltwater fishes. The freshwater fish values are used throughout the PEIS. Temperature and water quality effects are only indirectly considered as the original sources are based on experiments over a range of conditions. Backup studies at nuclear plant sites indicate that the models in which these factors are used are conservative.

13.9.5.10 Radiation Doses in the Aquatic Food Chain (11, 14, 20, 21, 30, 37, 42, 45, 46, 52, 63, 64, 66, 68, 85)

The concentration of radioactive material in fishes in the Susquehanna River and the Chesapeak Bay should the discharge of processed water be permitted is estimated in Section 7.2. The bioaccumulation factors for fishes have taken into consideration for bioaccumulation in lower trophic levels. Actual bioaccumulation factors for lower trophic levels (invertebrates) are smaller than for fishes. Hence, the radioactive concentrations in fishes represent the maximum concentration that would be expected throughout the food chain. The assessment in the PEIS indicates that the concentrations of radioactive material and the radiation doses to fishes and other aquatic biota are very low. These doses to Susquehanna River fishes and Chesapeak Bay fishes will not result in detrimental effects to those fish populations nor should have an effect on the entire aquatic food chain or commercial fishery.

13.9.5.11 Effect of Radiation on Livestock (68)

The impacts from TMI on livestock and the results of the NRC analysis are published in NUREG-0378, "Investigations of Reported Plant and Animal Health Effects in the Three Mile Island Area." While many of the symptoms reported are characteristic of radiation sickness and of many other diseases, the necessary spectrum of symptoms that would establish a causal link between the reported problems and TMI was not in evidence. Considering the lack of systematic geographic pattern of reported problems related to the power plant, and that many of the problems were diagnosed as common occurrences in domestic and wild animals, the staff believes that no relationship can be established between the operation of TMI, the accidental releases of radioactivity and the reported health effects. The major source of contamination to milk is the possibility that radioiodine will be taken up by the cows and then delivered to the milk. The source of radioiodine has not existed since the latter part of May of 1979. Therefore, we do not believe that sampling farm milk along the transportation routes will be required.

13.9.5.12 Behavior of Sorbed Radionuclides in River and Bay (72)

Appendix T has been added to the PEIS. The behavior of radionuclides in the Susquehanna River and Chesapeake Bay is discussed in Sect. 7.2. The staff expects that the sorption of radioactivity by sediment will have only a minor effect on the ecology of the river and bay.

13.9.5.13 Concentration Factors in Water (72)

There is an aquatic concentration factor for radionuclides in fish. However, there is no concentration factor in water. The concentration factors for fish used by the NRC staff are described in Regulatory Guide 1.109 (bioaccumulation factor). The concentration factor for cesium is 2000 and for strontium is 30. Concentration of radionuclides in fish was estimated in the PEIS for the Susquehanna River and the Chesapeake Bay. Estimates of bioaccumulation factors for cesium and strontium based on measured values in water and fish of the Susquehanna River are presented in Section 7.2.5.4. These comparisons indicate that the Regulatory Guide 1.109 values are conservatively high.

13.9.6 Environmental Monitoring

13.9.6.1 Environmental Monitoring During the TMI-2 Accident (70)

Extensive environmental monitoring programs exited during the accident, including those conducted by the licensee, the NRC, the EPA and the DOE. Refer to NUREG-0558, "Population Dose and Health Impact of the Accident at the Three Mile Island Nuclear Station," for a detailed discussion of the monitoring program that was in existence around TMI at the time of the accident and the analysis that was used to determine the dose to the population surrounding TMI.

13.9.6.2 Environmental Surveillance Program (73, 74, 75)

The Executive Office of the President has designated the EPA the lead federal agency for conducting a comprehensive, long-term environmental radiation surveillance program. In addition to its own extensive program, EPA also coordinates all other environmental monitoring programs conducted by the licensee, the Department of Energy, the Commonwealth of Pennsylvania, the State of Maryland and the NRC. The diverse programs provide verification of the results and should be considered to be superior to one single program. This coordinated effort provides the most extensive monitoring around any nuclear power facility in the United States. In addition to the environmental monitoring programs, the licensee must continuously monitor all radioactive efluents from the facility. The inplant radioactive effluent monitoring program should detect any short-duration releases, and, coupled with the onsite meteorological monitoring program, enable the assessment of offsite dose consequences. The radiological and aquatic monitory programs are discussed in Section 11.

13.9.6.3 Effluents Monitoring and Offsite Dose Assessments Required (73)

The environmental monitoring programs are not the only programs keeping track of airborne releases into the environment during the cleanup. They are augmented by the plant effluent monitoring program required by Appendix A of 10 CFR Part 50, General Design Criterion 64, for the monitoring of radioactive releases. Appendix R of the PEIS, Proposed Additions to Technical Specifications for TMI-2 Cleanup Program, requires estimates of amounts and types of radioactivity to be released and reporting of amounts and types of radioactivity that were released.

The plant effluent releases are monitored prior to atmospheric dispersion or aquatic dilution. In addition to its effluent monitoring program, the licensee is required to estimate population and maximum individual doses to ensure compliance with offsite dose limits. Therefore, the environmental monitoring sensitivity limits need not allow for the detection of the very low environmental concentrations estimated in the PEIS for the cleanup operations.

13.9.6.4 Independent Environmental Monitoring (73)

In addition to the licensee's environmental monitoring program, there are several other environmental monitoring programs, including those conducted by the EPA and Commonwealth of Pennsylvania, which are independent of the interests of the licensee. The NRC issues a weekly status report that contains data on the radionuclide effluents and the results of environmental monitoring by the EPA environmental monitoring program.

13.9.6.5 Technical Specification Requirements on Offsite Dosimeters (3, 27)

A standard monitoring format is used for all new technical specifications. The new technical specifications require that there be two rings of dosimeters around a plant, one at the site boundary, and one at a distance of three to five miles. Each ring is comprised of 16 locations where dosimeters are placed, one in each of 16 compass sectors. This system has undergone extensive technical review, and we have concluded that, in conjunction with other monitoring systems which monitor releases of radioactivity at the plant, the public health is protected. The offsite dosimetry system at TMI has about twice the number of dosimeter locations as the standard one described here. The background radiation level in the vicinity of TMI has not changed significantly since the radiological survey of 1976 prior to the operation of Unit 2. The NRC issues a weekly TMI-2 status report. Any significant increases in environmental dosimeter results will be reported to the public.

13.9.6.6 Technical Specification Requirements on Offsite Water Monitoring (28, 74)

The technical specifications for TMI-2 require that Susquehanna River water samples be taken above and below the plant at a monthly and quarterly frequency. The licensee also has a sampling program for fish, aquatic plants, and aquatic sediments. This sampling, in addition to the sampling of all processed water, will assure that the public is protected from planned or unplanned discharges of radioactivity. For surface water samples, monthly gamma spectra analyses are required and quarterly tritium analyses are required. For drinking water samples, monthly gamma spectra analyses, monthly gross beta analyses, monthly I-131 analyses, and quarterly tritium analyses are required.

13.9.6.7 Environmental Sampling Program (64, 68, 74, 105)

The licensee has an extensive program to appropriately sample and analyze materials related to the food pathways, such as agricultural products, milk, water, and soil, as well as fish, aquatic plants, and aquatic sediments as part of the Environmental Monitoring Program. Details of this program can be found in Section 11 of the PEIS.

13.9.6.8 Monitoring of River Sediments (64)

The technical specifications for TMI-2 require that the Susquehanna River sediments and water be periodically sampled for radionuclides. The sediments are required to be sampled semi-annually at upstream and downstream locations. A gamma scan analysis is done on the sediment samples and the results are reported to NRC.

13.9.6.9 Duration of Monitoring Programs (56)

The environmental monitoring will be long-range and will last as long as necessary, certainly for the five- to seven-year duration of the cleanup. These programs are designed to monitor the human environment and are capable of detecting the first order impacts on the environment. In addition, cleanup workers are covered by a personnel dosimetry program to ensure that regulatory limits are not exceeded and that radiation protection programs to maintain radiation doses as low as reasonably achievable are effective.

13.9.6.10 Contingency-Surveillance Program (64)

The contingency-surveillance program is augmented by the onsite effluent monitoring program. For example, should an unplanned airborne release occur, the effluent monitors and the onsite meteorological monitors will be able to rapidly assess the offsite radiation levels prior to activation of the offsite contingency-surveillance program. In the event of a significant release into the river, the contingency program will sample and analyze the water on a timely basis. The routine weekly composite sample analysis is performed during normal cleanup operations.

13.9.6.11 Monitoring of Unplanned Releases (74, 75,)

The licensee is required to establish and implement a continuous onsite radioactive effluents monitoring program such that any release from the facility will be quantified. In the event of a short-term unplanned release, the onsite effluents monitors would detect the release and enable the following actions to be taken: (1) mitigation actions to stop the release, e.g., shut off exhaust or discharge, stop the source of leakage, halt cleanup operations, etc., (2) notification of NRC and EPA to enable activation of contingency environmental monitoring, if necessary, and (3) quantification of the release to the environment for the calculation of offsite radioactivity levels and dose rates. The existing environmental monitors would continuously monitor the radiation levels (e.g., TLD dosimeters, continuous air samplers, composite water samplers) such that subsequent measurements can be used to confirm the offsite dose level calculations.

13.9.6.12 Onsite Monitors to Augment Environmental Monitoring for Estimation of Maximum Offsite Doses (74)

Estimates of the maximum offsite dose from airborne releases will be based on readings of monitors at the point of effluent release at the facility and on meteorological data, such as wind speed and direction, at the onsite monitors. The stationary monitors placed offsite will serve to confirm those dose calculations. Mobile monitors for release tracking will be available in the event of any significant airborne release.

13.9.6.13 Expansion of the Federal Monitoring Program (34)

No proposal for release of the processed water has been made by the licensee. If such a proposal is made, and if the NRC decides to permit the release, the environmental monitoring program will receive additional review. The EPA, when notified of the decision, could institute and coordinate additional, appropriate environmental monitoring programs.

The Baltimore Regional Planning Council suggests that, if the processed water is released from TMI-2, the current Federal monitoring program should be expanded along the Susquehanna River and upper Chesapeake Bay. The Baltimore City Health Department also recommends that Baltimore and the State of Maryland be represented on the official monitoring team. NRC will forward these suggestions to the EPA, which has the responsibility for coordinating the offsite monitoring programs relative to TMI.

13.9.6.14 Monitoring Programs for Specific Cleanup Activities (79)

The environmental monitoring programs are comprehensive and extensive. In the event of any significant anticipated release, the programs would be implemented to concentrate on the specific event, if appropriate. For example, during the controlled Kr-85 release from the reactor building, mobile monitors were employed to track the airborne releases.

13.9.6.15 Tritium Monitoring if Processed Water Is Disposed by Evaporation (28)

The method for disposing of processed water from the cleanup operation has not yet been decided. If forced or natural evaporation of water is used, NRC will consider whether or not tritium air monitoring will be required and determine the requirements of such a monitoring program to be incorporated into the technical specifications. In the event that planned and unplanned airborne tritium release occurs, an airborne tritium sampling and analysis program will be instituted, e.g., during the krypton purging operation, a molecular sieve sampler was operated at the Observation Center for collection of atmospheric moisture for tritium analysis.

13.9.6.16 Updating the Aerial Radiological Survey of 1976 (3)

The aerial radiological survey of 1976 was conducted by the Energy Research and Development Administration (ERDA) for the purpose of measuring regional radiation levels. This activity was unrelated to surveillance of radiological impacts from TMI. Present environmental monitoring programs around TMI are extensive and were specifically designed for the purpose of monitoring environmental impacts of operations at TMI. In addition, releases from the accident were essentially composed of noble gases and did not result in any deposition which could be detected by a further aerial survey at this time. Therefore, updating the ERDA survey of 1976 is unnecessary.

13.10 ACCIDENTS AND SEVERE NATURAL PHENOMENA

13.10.1 Accidents

13.10.1.1 Accidents Involving Human Error (22, 32, 50, 55, 59, 91, 92, 102, 120, 122)

It is recognized that during the cleanup operation accidents could occur due to either equipment failure or human error. For each of the processing alternatives and operations, the possibility of accidents occurring has been considered and the potential effects of these accidents on the health and safety of the public and workers have been examined.

13.10.1.2 Accident Effects during Unit 1 Operation (32, 60)

The staff has reviewed the ramifications of a TMI-2 type of accident at Unit 1 during the cleanup at Unit 2, and concluded that there should be no significant additional environmental impact in this regard. The following factors were taken into consideration: (1) the only significant releases from the plant during the TMI-2 accident were noble gases. The environmental impact would be no more severe either because of the cleanup operations or radioactive wastes temporarily stored onsite; (2) the licensee, in order to operate Unit 1, must provide demonstration of separation and/or isolation of the TMI-1 and -2 systems such as radioactive liquid transfer lines, fuel handling areas ventilation systems, and sample lines. Therefore, the cleanup should not affect the emergency operation during a TMI-1 accident; (3) TMI-1, if restarted, will be staffed with sufficient personnel such that there would be enough personnel available to backup and aid in managing the accident independent of TMI-2 cleanup personnel and (4) if necessary, the cleanup operation can be halted and delayed in the event of an accident at Unit 1. Since the cleanup of TMI-2 began soon after the accident of TMI-2, such a delay in cleanup of TMI-2 is not expected to be prolonged. The environmental impact of such a delay should not be significant.

Conversely, because of the separation and isolation of the TMI Units 1 and 2, there should be no significant impact on Unit 1 if a cleanup accident occurs at Unit 2. For example, if there is an accident in the Unit 2 reactor building during defueling that results in any significant airborne radioactivity in the building, the building ventilation will be shut down to prevent airborne releases. However, even if there is a significant airborne release, the Unit 1 control room ventilation could go into emergency mode and the safety of Unit 1 would not be compromised.

13.10.1.3 Potential Formation of Zirconium Hydride and Fire Hazards (1, 70, 73)

The NRC staff has reviewed the potential for fire hazards due to any zirconium hydride that may have formed and concluded that it is unlikely that a zirconium hydride ignition would occur. The staff bases its conclusion on the following considerations: (1) Review of data on core condition indicates that there should not be significant amounts of zirconium hydride present in the reactor vessel. (2) Only finely divided zirconium hydride, in a powder form, when exposed to air (oxygen) would be pyrophoric. The presence of hydrided Zircaloy cladding in the powdered state would be readily identified by visual inspection with underwater TV montior during core inspection (prior to any defueling operation). Precautionary actions will be taken if such presence is identified to ensure that no fire hazard would occur. (3) Defueling operation will be performed with water coverage. Zirconium hydride would not ignite under water.

13.10.1.4 Basis for River Flow Rate in Storage Tank Rupture Accident (21)

The flow rate of 10,000 cfs was estimated from the rating curve at Goldsboro gage for an elevation of 279 ft MSL and refers to the East channel.[1] This assumed flow rate was used to estimate the consequences of a storage tank rupture and subsequent release of the processed water into the East Channel with subsequent dilution.

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[1] Letter (TLL 029) from R. F. Wilson, Met-Ed Co., to J. T. Collins, U. S. Nuclear Regulatory Commission, Subject: TMI-2 Processed Water Storage Tanks, January 24, 1980.

13.10.1.5 Analysis of Potential Leakage of Reactor Building Sump (50, 52, 54, 75, 100, 103, 120)

Details of the analyses of leakage of reactor building sump water are presented in Appendix V which deals strictly with the consequences of a non-mechanistic accidental leakage to the ground water. The appendix does not however, attempt to quantify the probability of such an event because it is not considered to be credible.

Numerical values of peak concentrations in the river as a result of the postulated leakage of reactor building sump water are also provided.

The outcome of the analysis demonstrates that even for the severe assumptions chosen, the resulting concentrations in the Susquehanna River are well below 10 CFR Part 20 standards for unrestricted uses. In addition, mitigative measures could be taken in the event of a significant loss of water from the sump.

13.10.1.6 Response of NRC to Incidents (102)

The NRC maintains a 24-hour incident response center. In the event of an incident, the licensee will notify the incident response center and appropriate action by the NRC staff will be taken.

13.10.1.7 Evacuation Routes (86, 92)

In determining the potential need for rapid evacuation of the region around TMI in the event of an accident during cleanup, bounding case credible accidents were considered for all cleanup alternatives. Estimates of the offsite dose consequences of these postulated accidents indicate that the need for rapid evacuation would not arise. Without evacuation, the exposure to the maximum exposed individual offsite, for the worst credible accident, would still be a small fraction of existing criteria for potential radiation exposures that would require evacuation.

13.10.1.8 Population Dose Estimation in Accident Analysis (103)

For each of the cleaning alternatives, bounding case accidents were postulated and analyzed for their potential dose impact to the public. The results calculated for the maximal exposed person indicate that the potential radiation exposure dose is relatively small in comparison with the accident dose calculated for reactor operations. Under this circumstance, and in view of the fact that the probabilities for the bounding case postulated accidents are very small, the calculation of population exposures (person-rem) is not warranted.

13.10.1.9 Accidents Involving Impact of Airplane Crash (32, 50, 51, 120)

The reactor building (RB) and the auxiliary and fuel handling building (AFHB) are designed to withstand the impact of an airplane crash. (The types of aircraft using the Harrisburg International Airport have been taken into consideration.) The consequences of an airplane crash with respect to the radioactive wastes that must be stored temporarily on-site outside of the RB and AFHB, were also considered. In any event, the probability of an aircraft striking the site is less than 10^{-7} per year, which is generally accepted as the limit beyond which events are not likely to occur.

13.10.2 Severe Natural Phenomena

13.10.2.1 Potential Impact of Flooding (32, 67, 96, 103, 109, 120, 123)

As noted in Section 10.5, the potential for releases due to flooding is small for the following reasons: First, flood-control dike protection is provided for radioactive waste storage cells. The river-water elevation has to exceed 304 ft MSL to flood over the cells. This is an unlikely event; the probability of recurrence interval was estimated by the staff to be greater than 2,000 years. (The flood following Hurricane Agnes in 1972 resulted in an elevation of the river to 300.5 ft MSL.) Second, the stored radioactive wastes are protected by double containment in sealed steel containers and sealed storage cells. Finally, the severe flooding that would top the station dike would most likely last less than 4 days. Mitigating actions could be taken to reduce the water level during that time. It is unlikely that this time interval would cause significant leaks. In addition, the auxiliary and fuel handling building and the reactor building would not be affected by the probable maximum flood since those buildings are fully protected against entry of floodwaters.

13.10.2.2 Basis for the Probable Maximum Flood (22, 55)

Flooding is discussed for the case of the Probable Maximum Flood (PMF), which is larger than the Design Basis Flood (DBF), which in turn is larger than the largest historical flood. The PMF is the largest flood for which safety-related equipment is protected. The DBF is the maximum flood during which a plant may continue normal operation. The analytical basis for the PMF is given in detail in the Three Mile Island Unit 2 Final Safety Analysis Report. The PMF is based on very conservative estimates by the Corps of Engineers of Probable Maximum Precipitation[1] and runoff characteristics.[2] The flow at Three Mile Island for the PMF is 1,650,000 CFS with an elevation at the north end of the island of somewhat less than 310 ft, and 305 ft at the south end of the island. It should be emphasized that this is a very conservative flood, having a recurrence interval well in excess of 1000 years. This very low recurrence interval, combined with the double containment, make it extremely unlikely that any releases will occur due to flooding of the Susquehanna River.

The 1972 flood associated with Hurricane Agnes was the worst recorded flood since 1786, when such record keeping began. The U. S. Corps of Engineers has determined that the recurrence interval of that type of flood is between 400 and 500 years. The Corps of Engineers also estimated that the 1972 flood had a flow of about 1,000,000 CFS. The DBF for the TMI plant is 1,100,000 CFS and would have a crest elevation of about 304 ft MSL at the north end of the island. The grade elevation on the site is approximately 304 ft. Dikes have been constructed at a 310-ft elevation on the north end of the island where flood elevation is highest, and at 304 ft at the south end.

13.10.2.3 Tornado Effects (50, 55)

An assessment of tornado effects and their consequences is given in Section 10.5.2. The probability of a design basis tornado is less than 10^{-7} per year, and the cleanup facilities and waste storage areas are sufficiently protected so that releases are highly unlikely.

13.10.2.4 Effects of Natural Disaster on Private Property (67)

As for the effects on private property (i.e., "homes") from a severe natural disaster (e.g., floods, earthquakes), any property destruction would probably only occur as a direct consequence of the natural phenomena itself. For natural disasters up to certain levels of severity (e.g., Design Basis Flood, Tornado or Earthquake), the leakage of radioactivity would be prevented by the design features reviewed in the Safety Evaluation for the TMI-2 facility. As noted in Section 10.5, potential consequences of a more severe disaster (e.g., Probable Maximum Flood) were also found to be minimal.

13.10.3 Safeguards Against Sabotage (100)

The relationship between sabatoge and accidents is difficult to assess in all but probabilistic terms. Given the required safeguards imposed pursuant to Commission Regulation, 10 CFR 73, its likelihood is regarded too remote to be properly assessed in evaluative terms.

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[1] Corps of Engineers, Hydrometeorological Report No. 40.
[2] Corps of Engineers, Susquehanna River Basin Study, Appendix D, June 1970.

13.11 SOCIOECONOMIC AND PSYCHOLOGICAL EFFECTS

13.11.1 Social and Economic Impacts

13.11.1.1 Scope of the Staff's Socioeconomic Analysis (5, 32, 51, 64, 75, 86, 100)

The staff agrees that Cumberland County should have been considered in Section 3.1.6 because of its proximity to and heightened awareness of TMI. This has been done in revising the section for the PEIS. However, potential impacts to the New Cumberland Army Depot and the Mechanicsburg Navy Depot are not discussed because the staff was unable to develop a credible model linking postulated accidents during decontamination of TMI-2 with continued operation of these military facilities. Similarly, the only accidents which the staff has been able to postulate would not result in significant radiation doses requiring hospitalization; thus, the discussion in the draft statement on hospital care has been omitted from this statement.

In developing its environmental impact analysis the staff must reach a balance between the socioeconomic costs and benefits of analyzing all, even trivial, issues. Clearly, an analysis of all issues, because of its length, would be unreadable and thus counterproductive to public disclosure. The balance in the staff's analysis is achieved by considering the scoping process required under current CEQ regulations and by the application of professional judgment. The scoping process permits the staff to subjectively rank those concerns expressed by individuals in public meetings and written correspondence. Professional judgment augments the expression of public concerns by adding or focusing issues which may not be fully appreciated by the public. The staff's balancing process results in substantive, temporal, and spatial limits which describe the framework for the environmental impact analysis. Section 3.1.6.1 provides the rationale for the spatial limitation of the analysis. We believe this discussion, which has been rewritten for the PEIS, is adequate.

13.11.1.2 The Susquehanna River as a Community Water Supply (20, 130)

The statement on page 3-19 of the draft PEIS that "use of the Susquehanna River in community water supply systems is very limited" was in error. This has been corrected in Section 3.1.6, and additional users are now listed in Section 3.1.4.2.

13.11.1.3 Impact of Increased Construction Workforce (50)

Section 5.2.5.4 of this final Statement explains the reasons for not evaluating the impact of in-moving construction workers on the communities and the economy.

13.11.1.4 Effects of Decontamination Activities on Land and Inhabitants (75)

The comment was made that Section 3.1.6.2 (Impact Study Area) does not include effects of the cleanup operation. The writer apparently did not recognize that this section is intended to be purely descriptive. Analysis of the effects is in Section 10.6 of the draft statement. Analyses of socioeconomic impacts are found in each section of this final Statement where such discussions are warranted.

13.11.1.5 Property Value Effects (50, 131)

In general, the staff agrees with the Met-Ed comment that recent data indicating increases in property values between 1979 and 1980 are contrary to the statement in item 6 of Chapter 12 of the draft Statement, which says "reduced property values ... may occur." Section 3.1.7.2 of this PEIS contains information on this subject which was not available when the draft PEIS was written.

13.11.1.6 Potential Impact on the Marketability of Chesapeake Bay Shellfish and Finfish (11, 13, 16, 29, 30, 34, 41, 42, 48, 52, 61, 64, 66, 72, 100, 130)

The potential for adverse impact to Chesapeake Bay activities--commercial fishing, seafood consumption, recreation fishing, and waterfowl hunting--as a result of controlled or uncontrolled releases of tritiated water to the Susquehenna River is addressed in Section 7.3.5.4. The staff concluded that an in-depth study of the potential economic losses to Chesapeake Bay activities being undertaken by the State of Maryland should precede the decision regarding disposition of TMI-2 processed water. Section 3.1.6.1 contains descriptive information on the Chesapeake Bay economy.

13.11.2 Psychological Concerns

13.11.2.1 Terms Used (2, 15, 19)

Misuse of terms, particularly the word "phobia", was cited by several authorities. "Phobia" has been removed from the discussions of psychological concerns in the final PEIS. Other technical terms have also been examined for appropriateness and consistency by the staff and its consultants.

13.11.2.2 Analysis Incomplete (15, 32, 36, 44, 51, 61, 72, 84, 96, 99, 100, 106, 120, 124)

In response to comments suggesting that more detailed analysis be made of the psychological consequences of decontamination, the staff has expanded these discussions and placed them in the appropriate chapters of the final PEIS. However, suggested topics not within the scope of the PEIS, such as the issues relative to restarting Unit 1, have not been addressed.

13.11.2.3 Psychological Consequences (59, 67, 69, 85, 93, 100, 114, 115, 122, 123)

Many comments allege that the psychological consequences of the cleanup process are understated in the draft PEIS. In expanding the assessments of these consequences throughout chapters 5 through 9, the staff has reexamined its assessments and changed them where appropriate.

13.11.2.4 Impact of Accident Minimized (99, 109, 114, 128)

The psychological impact of the March 1979 accident at TMI-2 was minimized in Section 3.1.7, according to several comments. The expanded discussion in this Final PEIS (now in Section 3.1.6) examines the existing stress on the community, and includes recent research by Bromet (1980) and Hout et al., (1980).

13.11.2.5 Public Reaction Against Releases to the River (12, 52, 72, 112)

The staff acknowledges the many comments opposing the possible discharge of processed water to the river. Further analysis of the psychological effects associated with all the feasible water disposal alternatives is included in Section 7.2.

13.11.2.6 Long-Term Psychological Effects (67)

The staff's discussion of psychological impacts attributed to the TMI-2 accident has been revised to acknowledge two recent studies which suggest that measurable long-term psychological effects may occur (Sec. 3.6.3). However, no conclusive evidence will exist for many years.

13.12 FINANCIAL ASPECTS OF TMI-2 CLEANUP

13.12.1 Costs

13.12.1.1 Costs Not Available (59, 124, 125)

Cost figures were not included in the draft PEIS because they were not available in time. Relative costs estimates for the alternatives are given in the PEIS. Unlike an environmental statement on a proposed nuclear power plant, cost factors are secondary issues; the cleanup has to proceed for protection of public health and safety and of the environment. It is appropriate, however, from the standpoint of the National Environmental Policy Act, to consider the costs of proposed actions in relationship with available alternatives. Even in this context, however, consideration of costs are not dispositive.

13.12.1.2 Relative Costs of Alternatives (4, 11, 13, 16, 50, 52, 61, 64, 66, 70, 84, 92, 112, 117)

Relative cost estimates of the cleanup alternatives have been included in the PEIS. Financial costs, however, will not be a major factor in the Commission's decision on specific cleanup proposals as the mandated responsibilities of the NRC are to ensure public health and safety and ensure that the environmental impacts will be acceptable. Expenditures on the cleanup are not regulated by the NRC but are the responsibility of the licensee. Cost factors may well influence the licensee's cleanup proposals the licensee has the responsibility to make.

13.12.1.3 Decommissioning Cost (115)

The comment refers to a cost figure originated in Volume II of NUREG-0662, "Final Environmental Assessment for Decontamination of the Three Mile Island Unit 2 Reactor Building Atmosphere." The figure quoted originated from a study by Stanford Research Institute. It was included in Volume II of the NUREG for the purpose of reproducing all the inputs that were received on the venting question. The discussion of decommissioning costs was actually irrelevant to the topic of krypton venting. See Section 1.3 for an updated discussion of cleanup costs.

13.12.1.4 Cost-Benefit Analysis (32, 59, 64, 66, 105, 123)

Cost-benefit analysis, in the usual sense for nuclear power plant applications, would include the cost and benefit comparisons of the application versus alternative sources of power. In the case of the cleanup operations, the costs are discussed in terms of the relative worker radiation exposures associated with the alternatives, and the benefits would be in terms of the reduction or elimination of potential sources of radiation hazards to the public through cleanup and disposition of radioactive wastes.

13.12.2 Financial Responsibility

13.12.2.1 NRC Financial Responsibility (7, 20, 59, 67, 102, 115, 126, 130)

The NRC, as an independent regulatory body, is mandated to ensure public health and safety and the protection of the environment for licensed activities; it has no responsibility for the financial well-being of nuclear power, generally, and is not authorized to provide loans to utilities for the purpose of preventing bankruptcy. Additionally, it is not the function of the NRC to regulate the expenditures of Metropolitan Edison Company. The construction of the Submerged Demineralizer System (SDS) by the licensee is proceeding at his own risk, since the operation of this system will require the NRC approval and such decision would not be made pending evaluation of the alternatives and their environmental impacts in this final PEIS. Construction of the SDS has not prejudiced any NRC decision on which method is most appropriate. The NRC will evaluate the licensee's proposed cleanup activities in accordance with the mandate to ensure the health and safety of the public and to protect the environment.

13.12.2.2 Licensee Financial Responsibility (12, 38, 46, 66, 100, 101)

The licensee, Metropolitan Edison Company and its parent company, General Public Utilities are responsible for the expenses of the cleanup. However, the licensee is presently constrained by the Pennsylvania Public Utilities Commission from spending money from the rate payers on the cleanup of TMI-2. Should the licensee be financially unable to carry out the required cleanup, the NRC has the authority under existing laws to act on behalf of the federal government such that the cleanup can be continued without hazard to the general public.

13.12.2.3 Licensee Default (11, 27, 66, 67, 100, 103, 114, 115, 125)

The financial aspects of the licensee are not within the regulatory responsibilities of the NRC and it is not within the scope of the PEIS to discuss the financial impacts of cleanup on the licensee. The NRC Staff, however, has developed contingency plans in the event the licensee should go bankrupt and fail to carry out the cleanup responsibilities. Details of this contingency planning can be found in a report prepared by the NRC staff, "Potential Impact of Licensee Default on Cleanup of TMI-2," (NUREG-0689). A number of options are possible, including NRC's licensing another organization or another federal agency to clean up the facility or NRC's taking over the facility itself. However, additional funding, authorized and appropriated by the Congress, would be necessary if such circumstances arise. The question of who is the plant operator for the cleanup, however, does not affect the evaluations in the PEIS. The cleanup alternatives will remain the same even in the event the licensee should default on the cleanup obligations.

GLOSSARY

Absorbed dose--The energy imparted to matter by ionizing radiation.

Accident sludge--Sludge consisting of fine solid material which includes cement dust, dirt, resin beads, etc., which have settled out from liquid contaminated during the TMI accident.

Accident water:

(a) Water that existed in the TMI-2 auxiliary, fuel handling, and containment buildings, including the primary system, as of October 16, 1979, with the exception of water which as a result of decontamination operations becomes commingled with non-accident-generated water such that the commingled water has a tritium content of 0.025 µCi/mL or less before processing.

(b) Water that has a total activity of greater than 1 µCi/mL prior to processing except where such water is originally non-accident water and becomes contaminated by use in cleanup.

(c) Water that contains greater than 0.025 µCi/mL of tritium before processing.

Activation--The induction of radioactivity in material by irradiation with nuclear particles, usually neutrons produced by a nuclear reactor.

Activity--A measure of the rate at which a material is undergoing spontaneous nuclear transformations, giving off radiation. The special unit of activity is the curie (Ci).

AEC--Atomic Energy Commission, predecessor to the Nuclear Regulatory Commission and the Department of Energy.

AFHB--Auxiliary and fuel handling buildings.

AFR storage--Away-from-reactor storage; refers to spent nuclear fuel storage facilities located elsewhere than at a nuclear reactor site.

ALARA--As low as reasonably achievable.

Alpha radiation--An emission of particles (helium nuclei) from a material undergoing nuclear transformation; the particles have a nuclear mass number of four and a charge of plus two.

Alpha waste--Waste material contaminated by radionuclides that emit alpha particles; in particular, transuranic elements.

Anadromous fish--Fish that ascend freshwater streams from the sea to spawn.

Anions--Ions that are negatively charged.

Anticipated operational occurrence--Miscellaneous conditions or actions such as equipment failure, operator error, administrative error, that are expected to occur at a nuclear plant that are not of magnitude great enough to be considered an accident.

AP-citrox process--A two-step decontamination process consisting of a pretreatment step using an alkaline permanganate solution followed by a cleaning step using a solution of citric and oxalic acids.

APAC process--Decontamination process using an alkaline permanganate solution and an ammonium citrate solution.

APSR--See axial power shaping rods.

Aquifer--A subsurface geological formation containing sufficient saturated permeable material to transmit groundwater and to yield economically significant quantities of water to wells and springs.

Atomic number--The number of protons in the nucleus of an atom which is also equal to the number of electrons outside the nucleus. The collective term for atoms of the same atomic number is "element."

Axial power shaping rods (APSR)--Rods inserted into the core of a reactor to control the rate of energy released in localized regions of the core.

Background radiation--The level of radiation in an area which is produced by sources (mostly natural) other than the one of specific interest, e.g., cosmic radiation and radioactive elements in the atmosphere, building materials, the human body, and from the crust of earth. In the Harrisburg area, the background radiation level is about 116 mrem per year, not including any contribution from medical practice.

Benthic--Generally living on the bottom of a water body.

Beta particles--An electron or a positron (a particle with the same mass as an electron but with a positive charge rather than a negative one). Usually used to refer to a particle moving at a velocity high enough to produce ions. Beta particles are commonly emitted from the nuclei of atoms undergoing nuclear transformation. Also referred to as beta radiation.

Beta radiation--See beta particles.

Bioconcentration (bioaccumulation)--The process whereby an organic system selectively removes an element from its environment and accumulates that element in a higher concentration.

Biomass--The total weight (mass) of living and dead organisms present in an area, volume, or ecological system.

Biota--Plant and animal life.

Biotic--Related to living organisms.

Bitumen--An asphalt material (usually obtained from petroleum or coal-tar refining) that can be used in immobilization of radioactive wastes.

Bituminization--Immobilization of wastes by addition of bitumen.

Body burden--The amount of a specified radioactive material or the summation of the amounts of various radioactive materials present in an animal or human body at the time of interest.

Borated water storage tank (BWST)--A tank at TMI-2 intended for use in storing water containing boron. This tank now contains water that has been processed through the EPICOR II.

Boron--A neutron-absorbing element used in nuclear reactor systems to control criticality.

Breakthrough volume--The processing volume passed through an absorbent (ion-exchange system) that necessitates the replacement or regeneration of the ion-exchange resins.

Burial ground--An area specifically designed for the disposal of solid waste.

Burnable poison rods--Rods containing materials with high neutron-absorbing capability which are used in the core of a reactor to control reactivity. As the rods are irradiated, the amount of neutron-absorbing material is reduced. This reduction in neutron-absorbing material compensates for the buildup of fission products in the fuel which absorb neutrons, and a general reduction in the amount of fissionable material as the fuel is irradiated.

Burnup--The process of fissioning (consuming) heavy metal isotopes in nuclear fuels.

BWR--Boiling water reactor.

BWST--See borated water storage tank.

Calandria--A sealed vessel containing cooling tubes used in the core of some types of reactors.

CAN-DECON process--A proprietary dilute chemical decontamination process developed by London Nuclear Services, Ltd., and Ontario Hydro.

Capable fault--A fault in the earth's crust determined to be capable of causing an earthquake of a specific intensity on the basis of the geologic history of the fault.

Carnivorous--Meat eating.

Catadromous fish--Fish living in fresh water and migrating to salt water to spawn.

Cation--An ion with a positive charge.

CCST--Chemical cleaning solution tank.

CCTV--Closed-circuit television.

cfm--Cubic feet per minute.

CFR--Code of Federal Regulations.

cfs--cubic feet per second.

Chelating agent--An organic compound used to complex some metal ions to prevent them from precipitating in neutral or alkaline wash solutions.

Chemical decontamination--Use of solvents to dissolve or suspend radioactive contaminants.

Ci--See curie.

Cladding--See fuel rod cladding.

Cleanup--In reference to TMI-2, this term is used to mean decontamination, core removal (defueling), and waste disposal.

Cold leg--The section of PWR reactor cooling system piping through which the coolant flows from the heat exchanger to the core.

Coliform bacteria--Intestinal bacteria that are considered as indicators of possible water pollution by human or animal wastes.

Condensate--The liquid product formed by condensation of a vapor.

Containment (or reactor) building--The structure housing the nuclear reactor. The building is specially constructed of reinforced concrete and is designed to withstand internal pressure and external collisions. It also is fitted with gas-tight seals designed to contain radioactivity within the building and permit release of radioactive materials only under controlled conditions.

Containment sump--See reactor building sump.

Contamination--In this document generally used to mean the deposition, solution, or infiltration of radionuclides on or into an object, material, or area.

Control rods (CR)--An array of tubes that contain material that absorbs neutrons and is inserted into the core of a nuclear reactor to control or halt nuclear fission.

Controlled area--Any specific region of a nuclear plant into which entry by personnel is regulated by a physical barrier or administrative procedure.

Core support structure--A large basket-like component that fits into the reactor vessel and is used to hold the fuel elements and direct the reactor coolant to the bottom of the reactor vessel.

Core--The central portion of a nuclear reactor containing the fuel elements.

Corrosion products--Materials formed by chemical reaction of metals with the coolant in the reactor.

CP review--Construction permit review; the review proceedings conducted by the NRC prior to issuance of a permit to construct a nuclear power plant.

CR--See control rods.

CRDM--Control rod drive mechanism.

Critical--The condition in which an arrangement of fissionable material undergoes nuclear fission at a self-sustaining rate.

Criticality safety--The handling and storage of fissionable materials in such a manner that they will become critical only under desired, controlled conditions.

Crud--Corrosion products (principally oxides of iron, chromium, and nickel) circulating in or loosely deposited on the surfaces of the primary system.

Cryogenic processing--Low-temperature separation of gases.

CSB--Containment service building.

Cumulative occupational dose--The total radiation dose to workers; determined by multiplying the dose rate times the number of workers exposed times the length of exposure. This is expressed in terms of person-rem.

Curie (Ci)--The special unit of radioactivity. Activity is defined as the number of nuclear transformations occurring in a given quantity of material per unit time. One curie of radioactivity is 37 billion transformations per second.

D-rings--The shield enclosures around the steam generator compartments; they are so named because of their shape.

Daughter products--The nuclides formed by the radioactive disintegration of a first nuclide (parent).

Dead leg--A section of reactor cooling system piping connected in a manner such that it does not drain with the rest of the system or such that water does not circulate through it.

Decay chain--The sequence of radioactive disintegrations in succession from one nuclide to another until a stable daughter is reached.

Decay heat--Thermal energy produced by the decay of radioactive materials.

Decommissioning--The planned, orderly execution of steps to place a facility in a permanently nonoperable, safe condition.

Decontamination--The removal of radioactive material from a surface or from within another material.

Deionized water--Water from which ionized impurities have been removed.

Demineralizer systems--Processing systems in which synthetic ion-exchange materials are used to remove impurities from water.

Design basis earthquake--See operating basis earthquake.

Design response spectra--Plots of the intensities (amplitude) versus the frequencies of ground movement. Once a spectrum is determined, a nuclear reactor structure's response to that motion is computed by determining the extent to which it amplifies movement of the ground Response spectra are determined from geologic histories of ground motion.

Desorb--To remove materials that have been adsorped on another material.

Detritus--Loose organic material formed from decomposition of organisms.

Disintegrations per minute (dpm)--Obtained by multiplying number of disintegrations per second (dps) by 60.

Disintegrations per second (dps)--The number of radioactive decay events occurring per second in a given amount of material.

DOE--U.S. Department of Energy.

Dose commitment--The integrated dose that results unavoidably from an intake of radioactive material starting at the time of intake and continuing (at a decreasing dose rate) to later time (usually specified to be 50 years from intake).

Dose--A general term indicating the amount of energy absorbed from incident radiation by a unit mass of any material.

Dosimeter--Dose meter; an instrument that measures radiation dose.

dpm--See disintegrations per minute.

dps--See disintegrations per second.

Earthquake intensity--See modified Mercalli scale.

Eductor--An ejectorlike device used to mix or agitate fluids. Eductors are frequently used to pump fluids.

EPICOR I--The liquid radioactive waste processing system designed for use during routine operation of TMI to clean up liquids containing activity less than 1 microcurie per milliliter of water.

EPICOR II--A filtration and demineralizer system designed to process some of the liquid radioactive waste resulting from the TMI accident. The system can be used on liquid waste containing between 1 and 100 microcuries of radioactivity per milliliter of water.

Estuary--A semienclosed coastal body of water that has a free connection with the open sea and within which sea water is measurably diluted with fresh water.

Eutrophic--Pertaining to a shallow lake containing a high concentration of dissolved nutrients and having periods of oxygen deficiency.

Evaporator bottoms--The residue in an evaporator following evaporation of liquids.

Exclusive-use vehicles--Refers to vehicles used only to transport radioactive waste from a single shipper.

Exposure--The condition of being made subject to the action of radiation; also, frequently, the quantity of radiation received.

FES--Final Environmental Statement.

Filtrate--Material that has passed through a filter.

Fissile material--Material capable of undergoing fission following the absorption of thermal (essentially zero-energy) neutrons, e.g., U-238 and Pu-239.

Fission products--The nuclides formed by the division of a heavier nucleus, typically in a nuclear reactor. Isotopes of essentially all elements are produced by fission of fissile materials. Fission products are the main radioactive components of high-level radioactive wastes.

Fission--The spontaneous or induced disintegration of a heavy atom into two or more lighter atoms with an accompanying loss of mass which is converted into nuclear energy.

Fissionable--Material capable of undergoing fission when struck by neutrons of sufficiently high energy, e.g., U-238

Freon--Tradename for any of several liquid or gaseous fluorinated hydrocarbons.

FSAR--Final Safety Analysis Report.

FTC--See fuel transfer canal.

Fuel accountability--A system for determining the location of all nuclear fuel to prevent diversion for non-intended purposes.

Fuel assemblies--Metal grid structures containing arrays of nuclear fuel rods.

Fuel processing--Chemical and physical reprocessing of spent uranium- or thorium-based fuels for separation and recovery of uranium, thorium, and plutonium from the fission product wastes.

Fuel rod cladding--Metal material forming the exterior of a nuclear reactor fuel rod.

Fuel rod--One of many metal tubes containing uranium fuel for a nuclear reactor.

Fuel transfer canal (FTC)--A water-filled channel that connects the reactor core and the spent fuel storage pool.

Fuel--See nuclear fuel.

Gamma radiation--Electromagnetic radiation of high energy (and short wavelength), emitted by nuclei undergoing internal changes. Gamma (rays) have the highest energy and shortest wave length in the electromagnetic spectrum and are capable of penetrating several inches of a solid such as concrete.

Gamma scan--Use of radiation-detection equipment designed specifically to detect gamma radiation and measure its energy. This process can be used to evaluate the integrity of irradiated fuel elements.

gpm--gallons per minute.

GPU--General Public Utilities Co.

Groundwater--Water that exists or flows below the ground's surface (within the zone of saturation).

Half-life--The time required for half of a given radioactive substance to decay.

Hands-on work--Work requiring the presence of workers for physical manipulation of contaminated equipment or for actual decontamination activities.

HEPA filter--High efficiency particulate air filter.

High-density fixative--An adhesive plastic similar to that used for caulking compounds or roof patching compounds, but with a high-density filler added to increase radiation shielding.

High-frequency anchor point--A geologic term meaning the maximum intensity (amplitude) determined from the ground motion spectrum; used to compute the design response.

High-level waste (HLW)--Spent nuclear fuel or the radioactive materials extracted from spent fuel during reprocessing.

High-pressure water jet--A high-pressure, low-flow-rate water-jet spray system designed for use in removing surface contamination.

High-specific-activity wastes (HSAW)--Wastes having higher activities than wastes which are routinely generated at nuclear power plants and which are disposed of by routine shallow land burial techniques.

Holdback-carrier solutions--Solution containing inactive atoms which exchange with the radioactive species adhering to surfaces or equipment.

Hot cell--A heavily shielded work area designed to contain highly radioactive materials and including provisions for remote manipulations of such materials.

Hot leg--The section of PWR reactor cooling system piping through which the coolant flows from the reactor core to the heat exchanger.

Hot spots--Specific locations where radiation dose rates are significantly higher than in the general surroundings.

HPI--High-pressure injection system. This system is used to inject water into the reactor vessel in the event of a loss-of-coolant-accident.

HSAW--See High-Specific-Activity waste.

HTO--Tritiated water in which one of the two hydrogen atoms has been replaced by a tritium atom (see tritiated water).

Hydrogen control subsystem--A portion of the reactor building ventilation system designed to control the hydrogen concentration of the building atmosphere to within specified limits.

Ichthyoplankton--Fish larvae and eggs.

Immobilization--In this document, usually meant to refer to the fixing or solidification of radioactive wastes by any of several possible means (e.g., solidification in cement).

In-situ--In place.

Induced-service employment--Jobs created to fulfill the service needs of workers moving into the area. These jobs could be in the public sector (e.g., hiring of more firemen) or the private sector (e.g., hiring of more waitresses at a local cafe). In some cases, the jobs could be with local firms providing goods or services directly for the cleanup effort.

Internals--See reactor internals.

Invertebrates--Animals without backbones (such as insects and worms).

Ion exchange media--Resins or zeolite materials used in ion exchange processes.

Ion exchange--Here, a process for selectively removing a constituent from a waste stream by reversibly transferring ions between an insoluble solid and the waste stream.

Ion--An atom or molecule from which an electron has been removed (positive charged ion) or to which an electron has become attached (negative charged ion).

Ionization--The process by which a neutral atom or molecule acquires a positive or a negative charge by removal or attachment of an electron.

Ionizing Radiation--Any form of radiation that generates ions.

Irradiation--Exposure to radiation, e.g., by being placed near a radioactive source or in an X-ray beam.

Isotopes--Nuclides with the same atomic number but with different atomic masses therefore having the same chemical properties but different physical properties.

Krypton-85 (Kr-85)--A radioactive noble gas with a half-life of 10.7 years. This was the principal radioactive contaminant in the TMI-2 reactor building atmosphere.

LCF--Latent cancer fatalities.

Lead screws--Elements that connect the reactor control rods and axial power shaping rods to their respective drive mechanisms (stators).

LET--See linear energy transfer.

Letdown coolers--Heat exchangers used to cool reactor primary water before routine purification.

Linear energy transfer--A measure of the capacity of biological material to absorb ionizing radiation. Specifically, this is a measure of the amount of energy deposited per unit length along the track of a charged particle (a beta ray, for example) as it passes through an absorber.

LLW--See low-level waste.

Long-lived isotope--A radioactive nuclide that decays at such a slow rate that a quantity of it will exist for an extended period.

Low-activity wastes--Radioactive wastes that are similar to routinely generated wastes which are disposed of by routine shallow land burial operations.

Low-level waste (LLW)--All radioactive waste materials that are not high-level or transuranic waste. Most TMI-2 wastes will be of this type.

LSA boxes--Boxes designed to hold radioactive wastes classified as low specific activity by Department of Transportation Regulations.

Macrophytes--Macroscopic plants, especially in aquatic habitats.

Maximum credible earthquake--The greatest intensity earthquake reasonably expected in a given region based on current and historical seismicity and geologic structure.

Maximum permissible concentration (MPC)--The annual average concentration of a radionuclide in air or water to which an individual may be continuously exposed without exceeding an established limit of radiation dose. Intermittent exposures at greater than the MPC can occur without exceeding the dose standard.

MDA--See minimum detectable activity.

MDHRS--See mini decay heat removal system.

Mechanical decontamination--Use of mechanical means, such as sandblasting, to dislodge radioactive contaminants.

Meltdown--A state in the core of a reactor in which melting of fuel and internal reactor components occurs.

Mesotrophic--Refers to a lake characterized by a moderate supply of nutrients in the water.

Meteorological dispersion factor --A factor (seconds per cubic meter) that takes account of site-specific meteorological data in relating the concentration (Ci per cubic meter) of radioactive materials at a given location to a release rate (Ci/sec) of radioactive material at another location.

mgd--million gallons per day.

Microcurie (μCi)--Unit for measuring radioactivity. One microcurie is one one-millionth of a curie (1/1,000,000).

Milk juggers--Farmers who sell milk directly to consumers.

Millicurie (mCi)--Unit for radioactivity, one millicurie equals one one-thousandth (1/1000) of a curie.

Millirem (mrem)--One one-thousandth (1/1,000) of a rem (see rem).

Mini decay heat removal system (MDHRS)--A special forced-circulation cooling system installed since the TMI accident to cool the Unit 2 core.

Minimum detectable activity (MDA)--Minimum level of radioactivity detectable above background.

Moderator--A material such as ordinary water or graphite used in a reactor to slow down high-velocity neutrons, thus increasing the likelihood of fission.

Modified Mercalli (MM) scale--An arbitrary scale of earthquake intensity, ranging from I (detectable only by instruments) to XII (causing almost total destruction).

Molecular sieve--A chemical compound with a latticelike formation used to absorb or separate molecules. Usually a synthetic zeolite.

mrem--See millirem.

MSL--Mean sea level.

MW--Megawatts: unit of power equal to 1,000,000 watts.

MWe--Megawatts electric: unit of power denoting output of an electric power plant equal to one million watts of electricity.

MWHT--Miscellaneous waste holdup tank.

Neutron activation--The process of irradiating a material with neutrons so that stable atoms in the material are transformed into radioactive nuclides.

Neutron--A particle in or emitted from the nucleus of an atom which is electrically neutral and has a mass approximately equal to that of a proton (the nucleus of a stable hydrogen atom).

Neutron-absorber--A material, such as boron, that will readily absorb neutrons emitted during a nuclear chain reaction and thus make the neutrons unavailable to contribute to continuation of the chain reaction.

Noble gases--Inert gases that do not readily react chemically with other elements. These gases include helium, neon, krypton, xenon, and radon.

NRC--See Nuclear Regulatory Commission.

NRTS--National Reactor Test Station, now Idaho National Engineering Laboratory (INEL), near Idaho Falls, Idaho.

NS-1 process--A proprietary decontamination process developed by Dow Chemical Company.

NSSS--Nuclear steam supply system.

Nuclear fuel--Fissionable material inserted into a nuclear reactor. The basic fuel for most of the commercial nuclear reactors in the United States is uranium oxide.

Nuclear power plant--A facility that uses energy from a nuclear reactor to produce electricity.

Nuclear radiation--Particles and electromagnetic energy given off due to the transformation occurring in the nucleus of an atom.

Nuclear reactor--A device containing fissionable material in which a chain of fission events can be maintained and controlled to meet a particular purpose.

Nuclear Regulatory Commission (NRC)--U.S. agency responsible for the licensing, regulation, and inspection of commercial, test, and research nuclear reactors, as well as nuclear materials.

Nucleus--A positively charged particle, made up of neutrons and protons, situated in the center of an atom and surrounded by a cloud of negatively charged electrons.

Nuclide--A species of atom having a specific mass, atomic number, and nuclear energy state.

Occupational radiation exposure--The radiation exposure to which workers at a nuclear facility are subjected during the course of their work.

OL review--Operating license review; the review proceedings conducted by the NRC prior to issuance of an operating license for a nuclear power plant.

Operating basis earthquake--An earthquake of an intensity through which a nuclear plant is designed to remain functional.

OPG process--Decontamination process using a solution of oxalic acid, hydrogen peroxide, and gluconic acid.

Order of magnitude--A specification indicating a range or factor of about ten.

Oxalic-citrate-peroxide process--Decontamination process using a solution of oxalic acid, ammonium citrate, and hydrogen peroxide.

Oxidation--A chemical reaction that increases the oxygen content of a compound.

Particulates--Small particles.

PEIS--Programmatic Environmental Impact Statement.

Penetrating radiation--Forms of radiant energy that are capable of passing through significant thicknesses of solid material; these usually include gamma rays, x-rays, and neutrons.

Penetration R-626 Cutout--An existing, normally sealed, penetration through the outer wall of the TMI-2 reactor building through which instruments can be inserted.

Percolation--Gravity flow of groundwater through the pore spaces in rock or soil.

Permissible dose--The dose of ionizing radiation that, in the light of present knowledge, carries negligible probability of causing severe somatic injury or genetic effect.

Person-rem--The sum of the individual radiation doses (collective dose) received by members of a certain group or population. It may be calculated by multiplying the average dose per person by the number of persons. For example, a thousand people each exposed to one millirem (1/1000 rem) would have a collective dose of 1 person-rem.

pH--A measure of the relative acidity or alkalinity of a solution; a neutral solution has a pH of 7, acids have a pH of below 7, bases have a pH above 7.

Piscivorous--Fish eating.

Plankton--Small organisms passively floating in the water; includes phytoplankton (plants) and zooplankton (animals).

Plateout--The deposition of a substance from suspension or solution onto the internal surfaces of the vessels (e.g., pipes) containing the liquid. In this document, plateout refers specifically to thin layers of radionuclides that were deposited on all exposed building and equipment surfaces inside the reactor building and on the insides of pipes and tanks during and after the accident.

Plenum--See upper plenum assembly.

Polymer--A large molecule formed by the union of small or simple molecules.

Population dose--The summation of individual radiation doses received by all those exposed to the source or event being considered, and expressed as person-rem. The same as collective dose.

PORV--Pilot operated relief valve, located in a pipe leading out of the top of pressurizer and designed to open automatically when the primary system pressure exceeds a safe limit and to close automatically when the system pressure is back to normal.

Positive displacement pump--A pump in which a measured quantity of liquid is entrapped in a space, its pressure is raised, and then it is delivered.

Pressure vessel--See reactor pressure vessel.

Primary production--The formation of the food chain base through photosynthetic carbon fixation by plants and bacteria in water.

Primary system--See reactor cooling system.

Primary water--Water in (or from) the reactor coolant system.

Process solids--Wet solids in the forms of sludge, high-solids-content slurries, or granular materials generated during the accident or during subsequent treatment of accident water and decontamination liquids.

Processed water storage tanks (PWSTs)--Tanks intended for use in storing TMI water that has been subjected to decontamination processing.

Processed water--Contaminated water that has been treated to remove radionuclides (exclusive of tritium).

Productivity factor--A factor used (in this document) to indicate the expected efficiency of the decontamination workers, who must wear heavy protective clothing, respirators and other encumberances when they are working in a radiation field; usually expressed as a fraction of normal productivity. The productivity factor has been used in calculating the amount of time expected to accomplish given tasks.

PRTR--Plutonium Recycle Test Reactor at Hanford, Washington.

psi--Pounds per square inch; a measure of pressure.

PWR--Pressurized water reactor (TMI-2 is this type of reactor). The primary system is pressurized to about 2200 psi so that it will not boil as it is heated in the reactor core.

R & D--Research and development.

R--See roentgen.

Rad--A unit of absorbed dose of ionizing radiation. A dose of one rad results from the absorption of 100 ergs of energy per gram of absorbing material.

Radiation spectrum--The intensity of radiation expressed as a function of energy.

Radiation survey--The instrumental evaluation of an area or object in order to detect, identify, and quantify radioactive materials and radiation fields present.

Radiation zone--Area that contains radioactive materials in quantities significant enough to require control of personnel entry to the area.

Radiation--Energy in the form of electromagnetic rays (radiowaves, light, X-rays, gamma rays) or particles (electrons, neutrons, helium nuclei) sent out through space from atoms, molecules, or atomic nuclei as they undergo internal change or resulting from particles and electromagnetic radiation interactions with matter.

Radioactive decay--The spontaneous natural process by which an unstable radioactive nucleus releases energy or particles.

Radioactive waste--Waste materials (solids, liquids, or gases), contaminated with radionuclides.

Radioactivity--Product of radioactive decay of an unstable atom.

Radioiodines--Radioactive isotopes of iodine.

Radioisotopes--Radioactive isotope (see also radionuclide and isotope).

Radionuclide--An unstable nuclide that undergoes radioactive decay.

Radiotoxicity--The toxic or poisonous property released by ionizing radiation during radioactive decay.

Radwaste--See radioactive waste.

RB--See reactor building.

RCS--See reactor cooling (or coolant) system.

Reactor (nuclear)--A device in which a fission chain reaction can be initiated, maintained, and controlled.

Reactor building encroachment area--A staging area for decontamination operations established inside the reactor building, near the entry point.

Reactor building sump--The lowest part of the reactor building, designed to receive and hold, on a temporary basis, drainage and overflow.

Reactor building--The structure housing the nuclear reactor. The building is specially constructed of reinforced concrete and is designed to withstand internal pressure and external collisions. It also is fitted with gas-tight seals designed to contain all radioactivity within the building and permit release of radioactive materials only under controlled conditions.

Reactor cooling (or coolant) system (RCS)--This system, also known as the primary system, consists of the reactor piping, steam generators, reactor coolant pumps, pressurizer, letdown piping, instrument and sample pipelines, and other components (up to the first containment isolation valve) that routinely come in contact with the reactor cooling water.

Reactor internals--The various component parts and systems within the reactor pressure vessel.

Reactor pressure vessel (RPR)--The steel vessel containing the reactor core; also referred to simply as the reactor vessel or the pressure vessel.

Reactor pressure vessel head (RPVH)--The closure, or lid, on top of the vessel that contains the reactor core and associated equipment.

Reagents--Substances or solutions used to produce a chemical reaction.

Recirculating vacuum filter system (RVFS)--A water filtration system that is being used to aid in the removal of sludge from contaminated water in the AFHB.

Recriticality--Reinitiation of a self-sustaining nuclear chain reaction.

Release fraction--The fraction of the radioactive material in a vessel assumed to be released if the vessel is ruptured or otherwise breached.

Rem--A unit of dose equivalent which is proportional to the risk of biological injury.

Resin liners--Cylindrical metal vessels used to contain the resins and/or zeolites during purification of contaminated water by ion-exchange processes.

Resins--Solid or semisolid products of synthetic origin used in ion-exchange processes for purification of liquids.

Riparian--Living or located along the shoreline of a body of water (at the land-water interface).

Roentgen (R)--Unit of gamma or x-ray exposure in air. Energetic gamma rays which produced an exposure of 1 R would deliver a dose equivalent of approximately 1 rem to a person.

Routine Operational Liquid Wastes (ROLW)--Water that was initially nonradioactive, e.g., river water, which becomes contaminated by contact with low-levels of radioactive materials. This is not accident water as defined by the Agreement between City of Lancaster, NRC, and Met-Ed.

RPV--See reactor pressure vessel.

RPVH--See reactor pressure vessel head.

RVFS--See recirculating vacuum filter system.

Safe shutdown earthquake (SSE)--The greatest intensity earthquake which a nuclear plant must be designed to withstand and still be able to be shutdown and maintained in a safe condition.

SDS--See submerged demineralizer system.

Seal plate assembly--A large ring that is fastened between the reactor vessel flange and the fuel transfer canal floor. When this ring is installed it forms a watertight seal allowing the fuel transfer canal to be flooded.

Sediment--Solid material in the water that is not in solution, but is either distributed in the water or settled out of it.

Selective absorbtion process--A separation process whereby a liquid is used to selectively absorb (separate) a selected material (gas) from a source gas stream (air).

Sequestering agent--See chelating agent.

Sere--A temporary biological community that occurs during a successional sequence on a given site.

SFP--See spent fuel pool.

Shielding--A barrier of solid or liquid material (e.g., lead, concrete, or water) which reduces the intensity of radiation as it passes through and which can be used to protect personnel from the damaging effects of ionizing radiation.

Short-lived isotope--A radioactive nuclide that decays so rapidly that a given quantity is transformed into its daughter products within a short period (usually those with a half-life of days or less).

Sludge--Sludge in the case of TMI-2 refers to a mixture of fine solid material which includes particles of cement dust, dirt, and resin beads, etc. that have settled out from a suspension in the water.

Slurry--A free-flowing, pumpable suspension of fine solid material in liquid.

Smears--Dry or moistened filter papers that have been rubbed on areas of suspected radioactivity and then subjected to analysis to determine the type and approximate amount of removable radioactivity.

Solubilize--To make a material soluble or to increase its solubility.

Solvent--A liquid capable of dissolving a solid or another liquid.

Source term--The quantity of radioactive material, released by an accident or operation.

Spent fuel pool (SFP)--A water-filled pool designed specifically for the storage of spent nuclear fuel.

Spool piece--A short, flanged section of pipe used to connect two pipelines.

Stators--Elements that provide the motive force in lifting or lowering the reactor control rods and axial power shaping rods.

Storage--Accumulation for later retrieval and disposal. Storage can be performed for periods of several months (interim) up to 10-20 years (long term).

Subcritical--The state in a nuclear reactor when the rate of neutron loss exceeds the rate of neutron production.

Submerged demineralizer system (SDS)--A demineralizer system, similar to EPICOR II, that the licensee proposes to construct in the spent fuel pool and use to process accident water.

Sump water--Water that has accumulated in a sump.

Sump--The lowest part of a building designed to receive and hold, on a temporary basis, drainage and overflow.

Supernate--The clear liquid above a settled solid or precipitate.

Surfactants--Abbreviation for surface-active agent. Detergents or soaps which alter surface tension or interfacial tension between water and other liquids or solids; used for decontamination purposes.

Technical Specifications--Requirements which are part of the NRC Operating License for a nuclear power plant.

Temporary contamination control envelope--A barrier, such as a plastic tent, used to enclose a space in an airtight envelope and prevent airborne contamination from being carried outside the space.

Thermoluminescent detector (TLD)--A solid-state device used to measure radiation doses (see dosimeter).

TLD--See thermoluminescent detector.

Total-body dose--The radiation dose to the total body, including the bone and all organs, from both external and internal radionuclides.

Transuranic (TRU) waste--Wastes which contain or are contaminated by in excess of 10 nanocuries (one nanocurie = one-billionth of a curie) per gram of transuranic elements (with mass numbers higher than uranium).

Tritiated water--Water in which one or both hydrogen atoms have been replaced by a tritium atom.

Tritium (H-3)--A radioactive isotope of hydrogen, approximately three times heavier than the "normal" (most abundant) form. The half-life is 12.5 years.

Trophic structure--A characteristic feature of any ecosystem measured and described either in terms of the standing crop per unit area or energy fixed per unit area per unit time; often used broadly to refer to the various levels of a food chain.

TRU waste--See transuranic waste.

Turbidity--A measure of the degree to which sediments and other foreign matter are suspended in water (cloudiness).

Unrestricted areas--An area to which access is not controlled for purposes of protection of individuals from exposure to radiation or radioactive materials.

Upper plenum assembly--A large cylindrical plate that fits into the reactor vessel to guide the control rods and press down on the fuel assemblies in order to maintain proper alignment.

Viscous--Refers to liquids that are thick and not free flowing.

Vitrified wastes--Radioactive wastes immobilized, or solidified, in glass.

Wake-cavity effect--The region of turbulance immediately to the rear of a solid body, like a building, that is formed when wind currents flow over and around the object.

Whole-body dose--See total-body dose.

χ/Q--Relative concentration; a term representing the concentration of a pollutant being emitted to the atmosphere divided by the emission rate.

Zeolites--Any of various natural or synthesized silicates used to purify water.

Zircaloy--A zirconum-base alloy used as the cladding for fuel rods and for other reactor core hardware.

Zone of saturation--A subsurface zone in which all the interstices are filled with water under pressure greater than the atmosphere.